ORNL 1835
Special

AIRCRAFT REACTOR TEST

HAZARDS SUMMARY REPORT

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W. K. Ergen
A. P. Fraas
E. R. McQuilkin
J. L. Meem

OAK RIDGE NATIONAL LABORATORY
OPERATED BY
CARBIDE AND CARBON CHEMICALS COMPANY
A DIVISION OF UNION CARBIDE AND CARBON CORPORATION
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FOREWORD

The Atomic Energy Commission requires that a Reactor Hazards Summary Report be submitted and approved by the Advisory Committee on Reactor Safeguards prior to the operation of a new reactor or the modification of an existing reactor in order to determine, and thus assure, the safety of the Commission's various reactor projects. In accordance with USAEC-OR-8401, Reactor Safety Determination, this report describes the hazards that may conceivably be associated with the Aircraft Reactor Test. All possible types of hazards are described as well as the extent to which these hazards have been evaluated and considered in the design and proposed operation of the reactor.

ACKNOWLEDGMENTS

The bulk of this report was prepared by the authors, with the assistance of the staff members of the Aircraft Nuclear Propulsion Project who are associated with the Aircraft Reactor Test. In particular, W. R. Grimes and W. D. Manly provided the portions of the report pertaining to chemical and metallurgical problems, respectively, and E. R. Mann prepared the material on reactor controls. In addition, considerable assistance has been solicited from several groups outside the project, including Robert F. Myers and R. D. Purdy of the Oak Ridge Office of the U. S. Weather Bureau and T. J. Burnett of the ORNL Health Physics Division.

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AIRCRAFT REACTOR TEST

HAZARDS SUMMARY REPORT

1. INTRODUCTION AND SUMMARY

The successful completion of a program of experiments, including the Aircraft Reactor Experiment (ARE),[1] has demonstrated the high probability of producing militarily useful aircraft nuclear power plants employing reflector-moderated circulating-fuel reactors. Consequently, an accelerated program culminating in operation of the Aircraft Reactor Test (ART) is under way. In order to adhere to the compressed schedule of the accelerated program, it is essential that the Atomic Energy Commission approve the 7500 Area in Oak Ridge as the test site by February 15, 1955. This report summarizes the hazards associated with operating the contained 60-Mw reactor of the ART at the proposed Oak Ridge test site.

Descriptions are given of the reactor, reactor cell, test site, reactor controls, and operating plan, prior to presentation of the hazards considerations. The hazards are classified into three major categories: 1) accidents with an appreciable probability of occurring, 2) accidents causing rupture of the pressure shell, and 3) accidents causing rupture of the reactor cell. Category (1) accidents would involve minor difficulties in the integrity of the reactor system and would not result in injury to operating personnel or the surrounding population. Category (2) accidents, which are extreme nuclear excursions, might be caused by and can cause major breaks in the reactor assembly, but due to the presence of the reactor cell which would remain intact there would be no injury to operating personnel or other people. The causes of category (2) accidents are described in a general manner since it has been impossible to date to describe a specific series of events that would lead to an extreme nuclear excursion. In the total facility destruction, category (3), the reactor and the reactor cell would be ruptured to release the accumulated fission products, but this could only be accomplished by means of extremely clever distribution of large quantities of explosives by a saboteur or by a large aerial bomb. In the analysis the most severe case has been presumed; namely, that complete volatilization of all of the fuel would occur. This is highly improbable since any such accident would require a large amount of heat at a high temperature, and its occurrence without dispersion of much of the fuel in particulate form seems to be unlikely. In addition, quantities of other materials that would also be present would appreciably increase the total heat input required.

1. Wm. B. Cottrell (ed.), Reactor Program of the Aircraft Nuclear Propulsion Project, ORNL-1234 (June 2, 1952)

The reactor of the ART is to be a 60-Mw reflector-moderated circulating-fuel type whose basic design is suitable for reactors to be used in aircraft.[2] The size and weight of the reactor and shield will conform with aircraft requirements, and, insofar as possible in the limited time available, the design of the important components will be based on concepts satisfactory for airborne applications.

Operation of the ARE demonstrated that a high-temperature circulating-fuel reactor could be built and operated and that materials and machinery for satisfactory operation at elevated temperatures had been developed. It showed that the predicted large negative temperature coefficient of reactivity and the resultant self-regulatory characteristics of the reactor could be achieved. In addition, it was found that most of the Xe^{135} was removed from the fuel into the gas blanket space in the pump so that the steady-state concentration of the Xe^{135} was only 3% of the normal equilibrium value.

The new principle of design introduced in the ART consists of circulating the fuel through the reactor in a single, thick, annular passage and achieving the major portion of the moderation with a beryllium reflector which will also serve as an important portion of the shield. With the resulting reduction in shielding requirements it has been possible to design the ART reactor in such a way that the entire fuel system is contained within a sufficiently small, shielded volume to provide a low-weight shield and a useful aircraft reactor. The purpose of the Aircraft Reactor Test is to validate the methods of construction and the predicted operating characteristics of such a reflector-moderated circulating-fuel reactor.

A reactor power of 60 Mw was selected because it is approximately the power that must be reached to demonstrate that the engineering problems are solved and that the operating characteristics are satisfactory for the higher powered reactors to be used in high-altitude supersonic strategic bombers. In addition, a reactor with a power level in the 60-Mw range will provide sufficient power to fly radar picket ships, patrol bombers, and other desirable aircraft. For a power level above 60 Mw, the cost appears to be directly proportional to the power. Also it does appear that a thoroughly satisfactory 200-Mw reactor can be built more quickly by first building a 60-Mw reactor and then following it with a 200-Mw reactor in which the benefits of the experience gained with the 60-Mw reactor will have been incorporated.

The design of the ART envisions an essentially spherical reactor in which the beryllium moderator will be lumped in a central island and in an outer annulus (see Fig. 2.1, Sec. 2, this report). Two centrifugal pumps, arranged in parallel, will circulate the fuel downward between the inner beryllium island and the outer beryllium reflector and out of the bottom of the core. The fuel will then turn and flow upward through the heat exchanger region, which is around the spherical core. From the heat exchanger region the fuel will return to the pumps and will again be discharged downward into the core. The reactor heat will be transferred in the heat

2. A. P. Fraas and A. W. Savolainen, ORNL Aircraft Nuclear Power Plant Designs, ORNL-1721, May 1954 (issued Nov. 10, 1954).

exchanger from the fuel to the secondary coolant (NaK). The fuel will be one of a number of fluoride salt combinations which have been shown to have acceptable physical properties. In particular, the $NaF-ZrF_4-UF_4$ fuel mixture is known to be satisfactory. The mixture $NaF-KF-LiF-UF_4$ has some very desirable properties and is under intensive investigation.

Quite a variety of shielding arrangements has been considered for the ART. The most promising seems to be one functionally the same as that for an aircraft requiring a unit shield, namely, a shield designed to give 1 r/hr at 50 ft from the center of the reactor. Such a shield is not far from being both the lightest and the most compact that has been devised. It will make use of noncritical materials that are in good supply, and it will provide useful performance data on the effects on the radiation dose levels of the release of delayed neutrons and decay gammas in the heat exchanger, the generation of secondary gammas throughout the shield, etc. While the complication of detailed instrumentation within the shield does not appear warranted, it will be extremely worthwhile to obtain radiation dose level data at representative points around the periphery of the shield, particularly in the vicinity of the ducts and of the pump and expansion tank region.

Several arrangements have been considered as means for disposing of the heat generated in the reactor. The most promising of these is one that resembles a turbojet power plant in many respects. It will employ radiators essentially similar to those suitable for turbojet operation. Conventional axial flow blowers will be used to force cooling air through the radiators. This arrangement will be flexible and as inexpensive as any arrangement devised. It will give thermal capacities and fluid transit times essentially the same as those in a full-scale aircraft power plant. It will also give some very valuable experience with the operation of high-temperature liquid-to-air heat exchangers that embody features of construction and fabricating techniques suitable for aircraft use.

In an effort to minimize the liklihood of important troubles developing during the course of the test, an extensive series of component development tests has been initiated. These tests have been designed to establish sound techniques for the fabrication of pumps and heat exchangers and to provide detail design information on such factors as clearances, etc. The operating experience gained in the course of these tests should prove most helpful in minimizing operating troubles with the ART and in diagnosing such troubles as may develop. These component development tests include experiments with a hot (high-temperature) critical assembly which will consist of the pump, header tank, and core system envisioned for the full-scale reactor. The expensive pressure shell and heat exchanger will not be included in this hot critical experiment.

Experience with the ARE has indicated the advisability of building, in addition to the critical assembly, a complete reactor-pump-heat exchanger-pressure shell assembly for operation as a component test in the experimental engineering laboratory. In a structure as complex as this it is

felt that there will probably be a number of mechanical problems of construction. Rather than go to extreme and awkward lengths to try to correct these by reworking the first unit, it is to be built as expeditiously as possible and operated simply as a high-temperature component test with no fissionable material. The experience gained in fabricating and shakedown testing this first assembly should not only prove invaluable in construction and operation of the second assembly for use with fissionable material, but should actually lead to an earlier operating date for the ART.

As the design of the facility progressed it became apparent that the major hazards would be much less serious than had at first been presumed. Also, the use of circulating fuel with its high negative temperature coefficient gives a reactor in which a nuclear explosion seems almost out of the question. In order to operate the ART at Oak Ridge under the safest possible conditions the entire reactor system, with the exception of the NaK-to-air radiators, will be enclosed in a cell consisting of an inner tank within which the reactor assembly will be installed and an outer water-filled tank. The cell will thus provide a water-filled annulus around the reactor assembly. The very compact installation envisioned results in a very low investment in sodium and NaK (about 1/20 of that required for the KAPL-SIR reactor designed for the same power level). A relatively small amount of energy would be released by reactions involving the liquid metals, and therefore a correspondingly small-diameter cell can be used.

The cell with a water-filled annulus will be adequate to absorb the amounts of energy that could be released in an extreme reactor catastrophe. It will be impossible for a fragment ejected from the reactor assembly by an explosion to rupture the inner tank of the cell because the pressure shell surrounding the reactor has been deliberately designed to yield at a pressure of 1000 psi and the maximum velocity of a fragment ejected at this pressure would be substantially below that required to penetrate the cell wall.

The 60-Mw reactor test unit was designed originally to be operated at the National Reactor Testing Station (NRTS) at Arco, Idaho. It was envisioned that the reactor could be pretested at Oak Ridge and then shipped to NRTS for the nuclear tests. However, a survey disclosed that construction and operation at NRTS would require at least six months longer than at Oak Ridge. Delays would be occasioned by conducting a construction operation 2,000 miles away and any small difficulty that might arise in reactor operation would be likely to introduce a major delay if that difficulty were not foreseen and plans to cope with it made in advance. No delay would be occasioned by construction of the small reactor assembly cell for use at the Oak Ridge site, and approval is being requested from the Atomic Energy Commission for operation of the Aircraft Reactor Test in such a cell at the Oak Ridge site.

Design data for the ART are presented in Table 1.1.

TABLE 1.1 AIRCRAFT REACTOR TEST DESIGN DATA

Power	
Heat, maximum (kw)	60,000
Heat flux (Btu/hr/ft^2)	Heat transported out by circulating fuel
Power (max/avg)	2:1
Power density, maximum (kw/liter of core)	1400
Specific power (kw/kg of fissionable material in core)	4500
Power generated in reflector, kw	2040
Power generated in island, kw	600
Power generated in pressure shell, kw	210
Power generated in lead layer, kw	132
Power generated in water layer, kw	64
Materials	
Fuel	$NaF-ZrF-UF_4$, 50-46-4 mole % or $NaF-KF-LiF-UF_4$, 11-42-44-3 mole %
Fuel jacket	Inconel
Moderator	Beryllium
Reflector	Beryllium
Shield	Lead and borated water
Primary coolant	The circulating fuel
Reflector coolant	Sodium
Secondary coolant	NaK
Fuel System Properties	
Uranium enrichment (% U^{235})	93.4
Critical mass (kg of U^{235})	13.5
Total uranium inventory (kg of U^{235})	30
Consumption at maximum power (g/day)	80
Design lifetime (hr)	1000
Burnup in 1000 hr at maximum power (%)	11
Fuel volume in core (ft^3)	2.96
Total fuel volume (ft^3)	5.64
Neutron Flux Density (avg)	
Thermal, maximum (n/cm^2· sec)	4×10^{14}
Thermal, average (n/cm^2· sec)	2×10^{14}
Fast, maximum (n/cm^2·sec)	8×10^{14}
Fast, average (n/cm^2·sec)	4×10^{14}
Intermediate, average (n/cm^2·sec)	10×10^{14}

Control

Shim control	One rod of 5% $\Delta k/k$
Rate of withdrawal	3.3×10^{-4} $\Delta k/k$-sec
Temperature coefficient	-5.5×10^{-5} $(\Delta k/k)/^{\circ}F$

Circulating Fuel-Coolant Systems

Fuel in Core	Li Fuel	Zr Fuel
Maximum temperature, $^{\circ}F$	1,600	1,600
Temperature rise, $^{\circ}F$	400	400
Flow velocity, ft/sec	7	7
Reynolds number	170,000	85,000

Fuel-to-NaK Heat Exchanger	Li Fuel	Zr Fuel	NaK Coolant
Maximum temperature, $^{\circ}F$	1,600	1,600	1,500
Temperature drop (or rise), $^{\circ}F$	400	400	400
Pressure drop, psi	35	55	50
Flow rate, ft^3/sec	2.7	2.7	12.6
Velocity through the tube matrix, ft/sec	8	8	36
Reynolds number	4,600	2,300	180,000

Cooling System for NaK-Fuel Coolant	
Maximum air temperature, $^{\circ}F$	750
Ambient airflow through NaK radiators, cfm	300,000
Radiator air pressure drop, in H_2O	10
Blower power required (total for 24 blowers), hp	600
Total radiator inlet face area, ft^2	64

Cooling System for Moderator	
Maximum temperature of sodium, $^{\circ}F$	1200
Sodium temperature drop in heat exchanger, $^{\circ}F$	100
NaK temperature rise in heat exchanger, $^{\circ}F$	100
Pressure drop of sodium in heat exchanger, psi	7
Pressure drop of NaK in heat exchanger, psi	7
Flow rate of sodium through reflector, ft^3/sec	1.35
Flow rate of sodium through island and pressure shell, ft^3/sec	0.53
Flow velocity of sodium through reflector and island, ft/sec	30
Reynolds number of sodium in reflector and island	170,000

System Volumes and Pump Data	Li Fuel	Zr Fuel	Na Coolant	NaK Coolant
Number of pumps	2	2	2	4
Pumping head, ft	50	50	250	280
Flow per pump, gpm	600	600	430	1300
Pump speed, rpm	2850	2850	4300	125
Pump power per pump, hp	40	65	16	100

Dimensions

Core diameter (in.)	21
Island diameter (in.)	11
Fuel region thickness (in.)	4.5
Reflector thickness (in.)	12
Shield thickness, lead (in.)	7
Shield thickness, water (in.)	31

2. THE SHIELDED REACTOR ASSEMBLY

The reactor is to be of the circulating-fluoride-fuel, reflector-moderated type. It will employ sodium-cooled beryllium as the reflector-moderator material and is designed to operate at 60 Mw. The reactor assembly will include the pressure shell, reflector, fuel and sodium pumps, and heat exchanger assemblies. The basic design is shown in Fig. 2.1, a vertical section through the reactor. A series of concentric shells, each of which is a surface of revolution about the vertical axis, constitute the major portion of the assembly. The two inner shells surround the fuel region at the center (that is, the core of the reactor) and separate it from the beryllium island and the outer beryllium reflector. The fuel circulates downward and outward to the entrance of the spherical-shell heat exchanger that lies between the reflector shell and the main pressure shell. The fuel flows upward between the tubes in the heat exchanger into the two fuel pumps at the top. From the pumps, which operate in parallel, it is discharged inward to the top of the annular passage leading back to the reactor core. The fuel pumps are sump-type pumps with gas seals. A horizontal section through the pump volute region is shown in Fig. 2.2. A schematic diagram of the reactor system is shown in Fig. 2.3.

Reflector-Moderator Cooling System

The reflector will be cooled by sodium circulated by two pumps at the top of the reactor. The sodium will flow downward through passages in the beryllium and back upward through the annular space between the beryllium and the enclosing shells. The central beryllium island will be cooled in a similar manner, except that the sodium will leave the bottom of the island to be returned to the top of the reactor through cooling passages in the main pressure shell. The sodium will return to the pump inlets through small torrodial sodium-to-NaK heat exchangers around the outer periphery of the pump-expansion tank region. The sodium pump and heat exchanger subassemblies will be positioned on either side of the fuel pump volute region. The pipe from the sodium pump discharge will make a slip fit into the reflector sodium inlet tube. The leakage through this slip fit into the sodium return passage will simply recirculate with no penalty other than a small increase in the required pump capacity.

Pressure Shell

The Inconel pressure shell will constitute both the main structure of the reactor and a compact container for the fuel circuit. The design has been modified somewhat from that shown in Fig. 2.1 to make the shell continuous through the vicinity of the headers and thus give better continuity of stress flow and a minimum of welding. Blisters made of 1-in.-thick plate welded to the outer surface of the shell between the NaK pipes will serve both to reinforce that weakened region and to provide for sodium flow up through the shell. The inner liner assembly of the pressure shell will consist of a 0.75-in.-thick Inconel shell, a 0.125-in.-thick hot-pressed B_4C

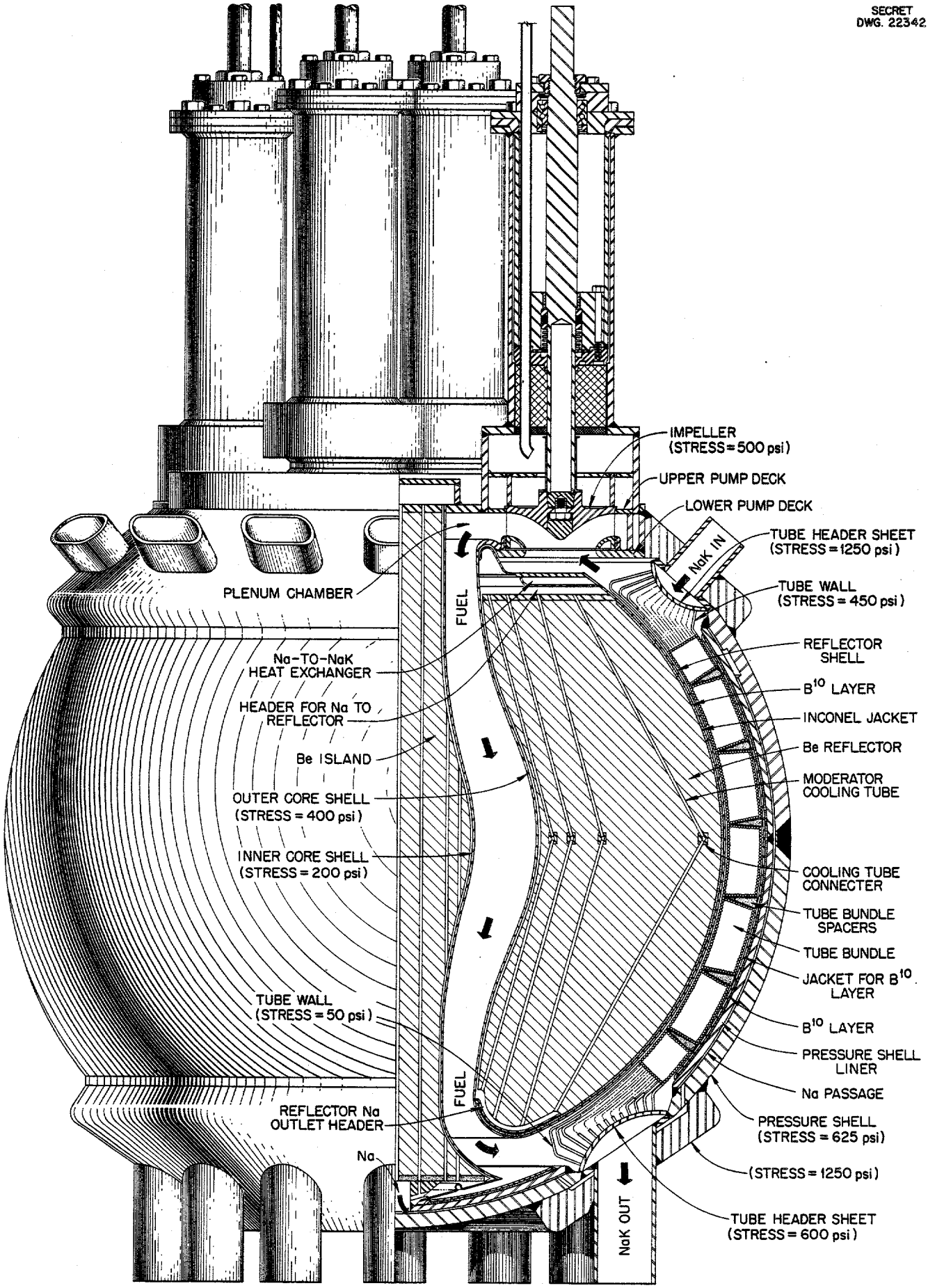

Fig. 2.1. 60-Mw Reflector-Moderated Reactor.

DWG E-17986B

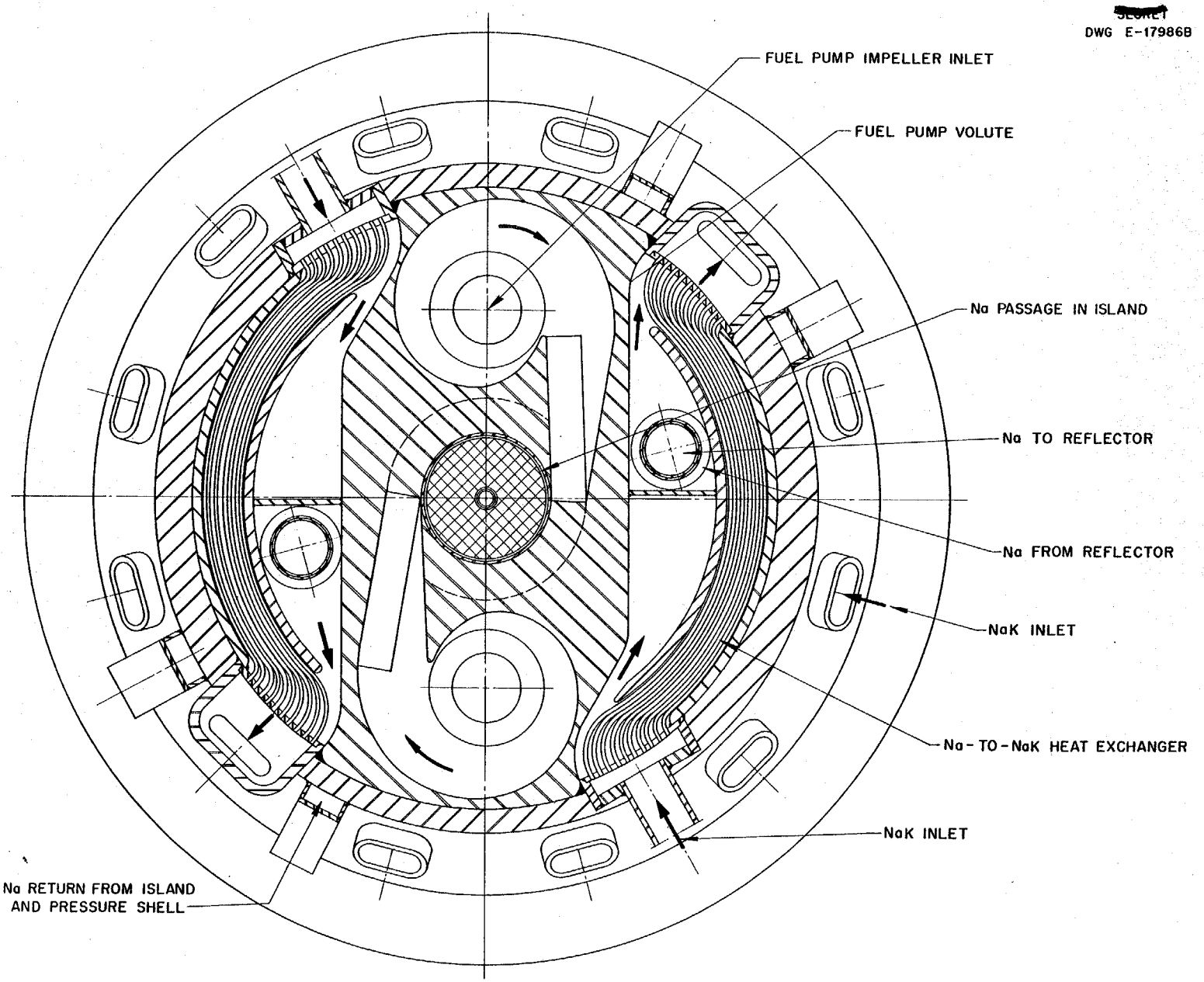

Fig. 2.2. Horizontal Section Through Pump Region of Reactor.

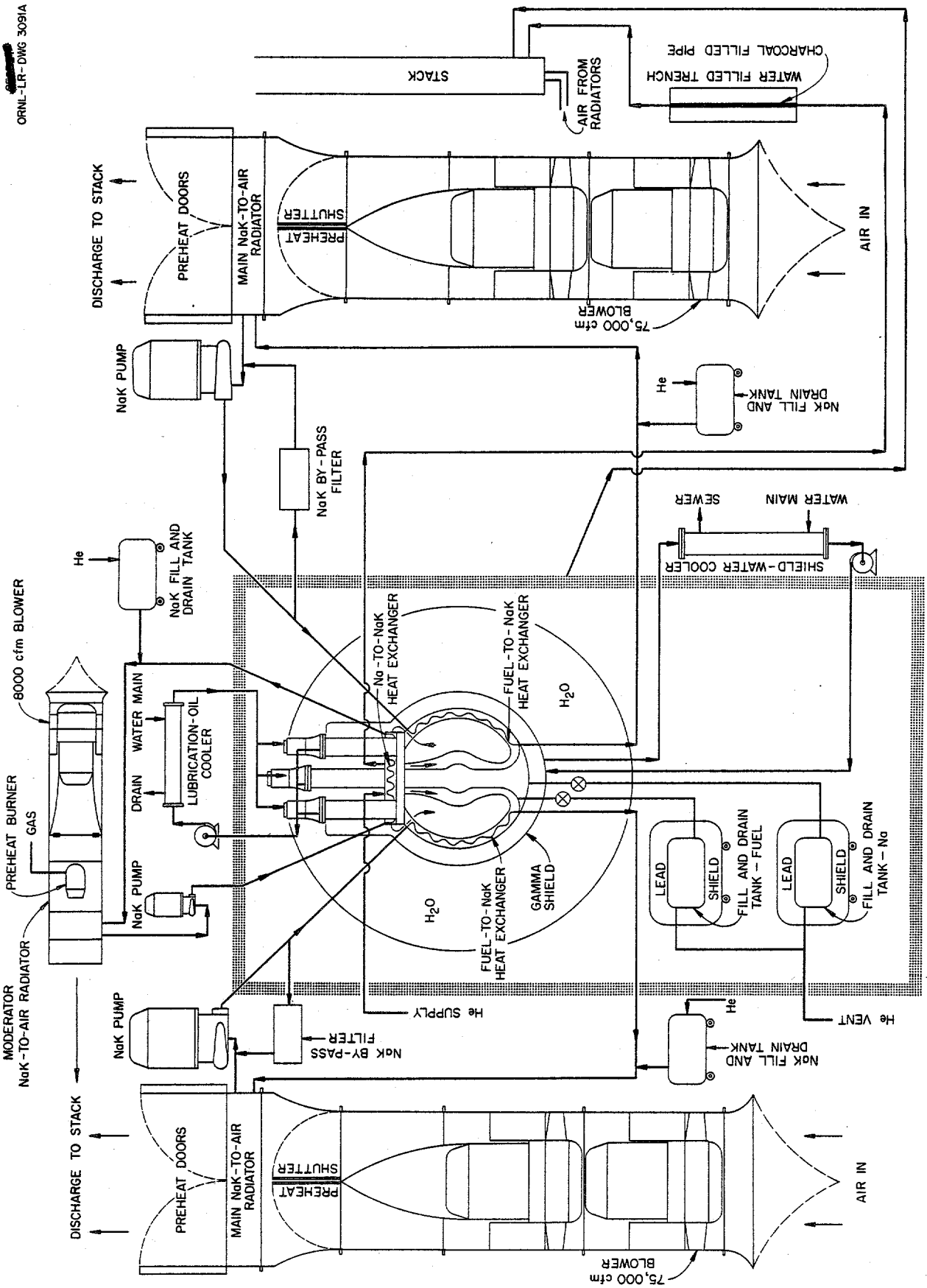

Fig. 2.3. Schematic of Aircraft Reactor Test.

layer, and an inner 0.062-in.-thick Inconel can. Heat generated in the pressure shell by the absorption of gammas from the fuel will be removed by sodium flowing between the outer surface of the liner and the inner surface of the pressure shell so that the pressure shell temperature will be held close to 1200°F. A 0.062-in.-thick gap will provide ample flow passage area for sodium to flow upward from the bottom of the island to the top of the pressure shell. Transfer tubes there will direct the sodium from the outer surface of the pressure shell to the sodium-to-NaK heat exchanger inlets. The hot-pressed boron carbide blocks will be diamond-shaped, with 60-deg angles at their vertexes, and they will have rabbeted edges. This design makes it possible to cover a spherical surface with a single block size and shape. To facilitate welding in the final assembly, the pressure shell will be split circumferentially at both 1°S and 35°N latitudes. By splitting the liner at 1°S latitude, it will be easily accessible for welding, and the upper portion of the main shell can be lowered into place for the final welding operations.

Heat Exchanger

The spherical-shell fuel-to-NaK heat exchanger, which makes possible the compact layout of the reactor-heat exchanger assembly, is based on the use of tube bundles curved in such a way that the tube spacing will be uniform, irrespective of latitude.[1] The individual tube bundles will terminate in headers that resemble shower heads. This arrangement will facilitate assembly because a large number of small tube-to-header assemblies can be made leak-tight much more easily than one large unit. Furthermore, these tube bundles will give a rugged flexible construction that will resemble steel cable and will be admirably adapted to service in which large amounts of differential thermal expansion must be expected.

Pumps

Two fuel pumps and two sodium pumps are located at the top of the reactor. These pumps are similar, but the fuel pumps have a larger flow capacity. In addition to pumping, the fuel pump will perform several other functions. Most of the xenon and krypton and probably some of the other fission-product poisons will be removed from the fluoride mixture by scrubbing it with helium as it is swirled and agitated in the expansion tank. The high swirl rate in the expansion tank is also desirable in that the centrifugal field will keep the free surface of the fuel reasonably stable in maneuvers or in "bumpy" flight. The expansion chamber will also serve as a mixing chamber for the addition of high-uranium-content fuel to the main fuel stream to enrich the mixture and to compensate for burnup. Fluid will be scooped from the vortex in the expansion tank and directed into the centrifuge cups on the backs of the impellers. Since considerably more fuel will be scooped from the vortex than can be handled by the centrifuge, the excess will be directed upward into the swirl pump where it will be accelerated

1. A. P. Fraas and M. E. LaVerne, Heat Exchanger Design Charts, ORNL-1330 (Dec. 7, 1952).

and returned to the expansion tank. There it will help maintain a high rotational velocity. A slinger located on the pump shaft above the swirl pump will prevent fuel from splashing up into the annulus around the shaft below the seal.

A number of other special features have been included in the pump design to adapt it to the full-scale reactor shield. The pump has been designed so that it can be removed or installed as a subassembly with the impeller, shaft, seal, and bearings in a single compact unit. This assembly will fit into the bore of a cylindrical casing welded to the top of the reactor pressure shell. A 3-in. layer of uranium just above a 1/2-in. layer of B_4C around the lower part of the impeller shaft will be at the same level as the reactor gamma shield just outside the pressure shell. The space between the bearings will be filled with oil to avoid a gap in the neutron shield. The pumps will be powered by d-c electric motors in order to provide good speed control.

Shield

The shield for the reactor has some characteristics that are peculiar to this particular reactor configuration. The thick reflector was selected on the basis of shielding consideration. The two major reasons for using a thick reflector are that a reflector about 12 in. thick followed by a layer of boron-bearing material will attenuate the neutron flux to the point where the secondary gamma flux can be reduced to a value about equal to that of the core gamma radiation. This thickness will also reduce the neutron leakage flux from the reflector into the heat exchanger to the level of that from the delayed neutrons that will appear in the heat exchanger from the circulating fuel. An additional advantage of the thick reflector is that 99% of the energy developed in the core will appear as heat in the high-temperature zone included by the pressure shell. This means that very little of the energy produced by the reactor must be disposed of with a parasitic cooling system at a low temperature level. The material in the spherical-shell intermediate heat exchanger is about 70% as effective as water for the removal of fast neutrons; so it too is of value from the shielding standpoint. The delayed neutrons from the circulating fuel in the heat exchanger region might appear to pose a serious handicap. However, these will have an attenuation length much shorter than the corresponding attenuation length for radiation from the core. Thus, from the outer surface of the shield, the intermediate heat exchanger will appear as a much less intense source of neutrons than the more deeply buried reactor core. The fission-product decay gammas from the heat exchanger will be of about the same importance as secondary gammas from the beryllium and the pressure shell.

Thermal insulation 0.5 in. thick separates the hot reactor pressure shell from the gamma shield, which is a layer of lead about 7 in. thick. The lead, in turn, is surrounded by a 31-in.-thick region of borated water. The slightly pressurized water shield is to be contained in shaped rubber bags similar to fitted aircraft fuel tanks. Cooling of the lead shield will be effected by

circulation of water through coils embedded in the outermost portion of the lead and through auxiliary coolers. The borated water shield will be cooled by thermal convection of the atmosphere in the reactor assembly cell.

Assembly and Testing

As each component of the reactor is constructed it will be cleaned and leak tested in the Y-12 area. The whole assembly is designed so that leak testing can be carried out as the components are added, piece by piece. Mass spectrographic techniques will be used for leak testing. The completed assembly of circulating-fuel and moderator-cooling systems can be leak tested before it is moved to the site. All thermocouples and other instrumentation will be checked as the assembly proceeds.

Assembly of the radiators, blowers, NaK pumps, fill and drain tanks, and other auxiliary equipment will proceed concurrently with the assembly of the reactor at Building 7503. Cleaning and leak testing procedures will be the same as those for the reactor. All instrumentation in the building will be checked out, as far as possible, prior to installation of the reactor package.

The reactor shield assembly (without water in the shield) will be moved as a package from the Y-12 area to the 7503 building. After all connections have been made, final leak testing will be carried out.

The NaK system will be filled with NaK and, with the heat barrier doors closed, the NaK pumps will be started. A heat input of approximately 300 kw will be attained by circulation of the NaK, and this energy in the NaK will be used to preheat the reactor. The presence of leaks, if any, from the NaK system to the fuel system will be determined at this time by a flame photometer. In addition to the leak check, the circulation of NaK in the system will permit the checkout of all instrumentation and the determination of the system characteristics. After circulation for several hours, the NaK will be sampled and analyzed for oxygen content. If the oxygen content is within the capacity of the purification system, the NaK will be left in the system for the remainder of the test; otherwise it will have to be replaced. If the NaK is to be replaced it will be dumped hot in order to carry the oxide with it.

The sodium system for cooling the beryllium will be filled next, and, finally, the fuel system will be filled with a barren fluoride mixture. Circulation of the fluids in these circuits will permit final cleaning of the systems, operational checks of instrumentation, and a determination of the system operating characteristics. As with the NaK, samples will be taken and analyzed for purity. The barren carrier will then be drained from the fuel system and replaced with a fluoride mixture containing 80% of the required uranium (as determined from the critical experiment).

3. THE TEST FACILITY

Site

The ARE facility, ORNL Building 7503, as noted in the ARE Hazards Summary Report,[1] is located at a site 0.75 mile southeast of the center of the present ORNL area and about 0.24 mile northeast of the Homogeneous Reactor Test (HRT) facility. This ARE location is near the center of Melton Valley which is approximately 4 miles long and 0.5 mile wide. With the exception of the nearby HRT, this valley is unoccupied. Between the ARE site (elevation 840 ft) and Bethel Valley, which contains ORNL (elevation 820 ft), is Haw Ridge which averages 980 ft in elevation. Within a radius of 1.9 miles, all the land is owned by the AEC and is already a security-controlled area. Within a radius of 2.3 miles there is approximately 0.3 square miles of farm land that is not AEC owned or controlled. Additional information on the surrounding area and the natural characteristics of the site is presented in Appendix A.

Building

To modify the ARE Building to accommodate the ART, it is planned that an addition to the south end will be constructed to effect a 64-ft extension of the present 105-ft long building. The ART shielded reactor assembly and its container will be installed within this addition to the ARE Building. Such an arrangement will permit the use of ARE services and facilities that exist in this installation which has now served the purpose for which it was erected. For example, items such as the control room, offices, change rooms, toilets, storage area, water supply, power supply, portions of experimental test pits, access roads, security fencing, and security lighting are available for incorporation in ART plans. Fig. 3.1 shows the preliminary design of the facility in perspective.

The plan and elevation drawings of the ART facility are shown in Figs. 3.2 and 3.3. The floor level of the addition will be at the ARE basement floor grade (ground level at this end of the building), and the cell for housing the reactor assembly will be sunk in the floor up to 3 ft below the bolting-flange level. The reactor cell will be located in approximately the center of the 42-ft wide by 64-ft long high-bay extension and directly in line with the ARE experimental bay. The reactor assembly will be positioned so that the top of the shield will be at the building floor elevation.

The south wall of the ARE experimental bay will be removed and the overhead crane facility will be revised from 10-to-20-ton capacity to permit use of the experimental pits for installation of auxiliary equipment

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1. J. H. Buck and W. B. Cottrell, ARE Hazards Summary Report, ORNL-1407, (1952).

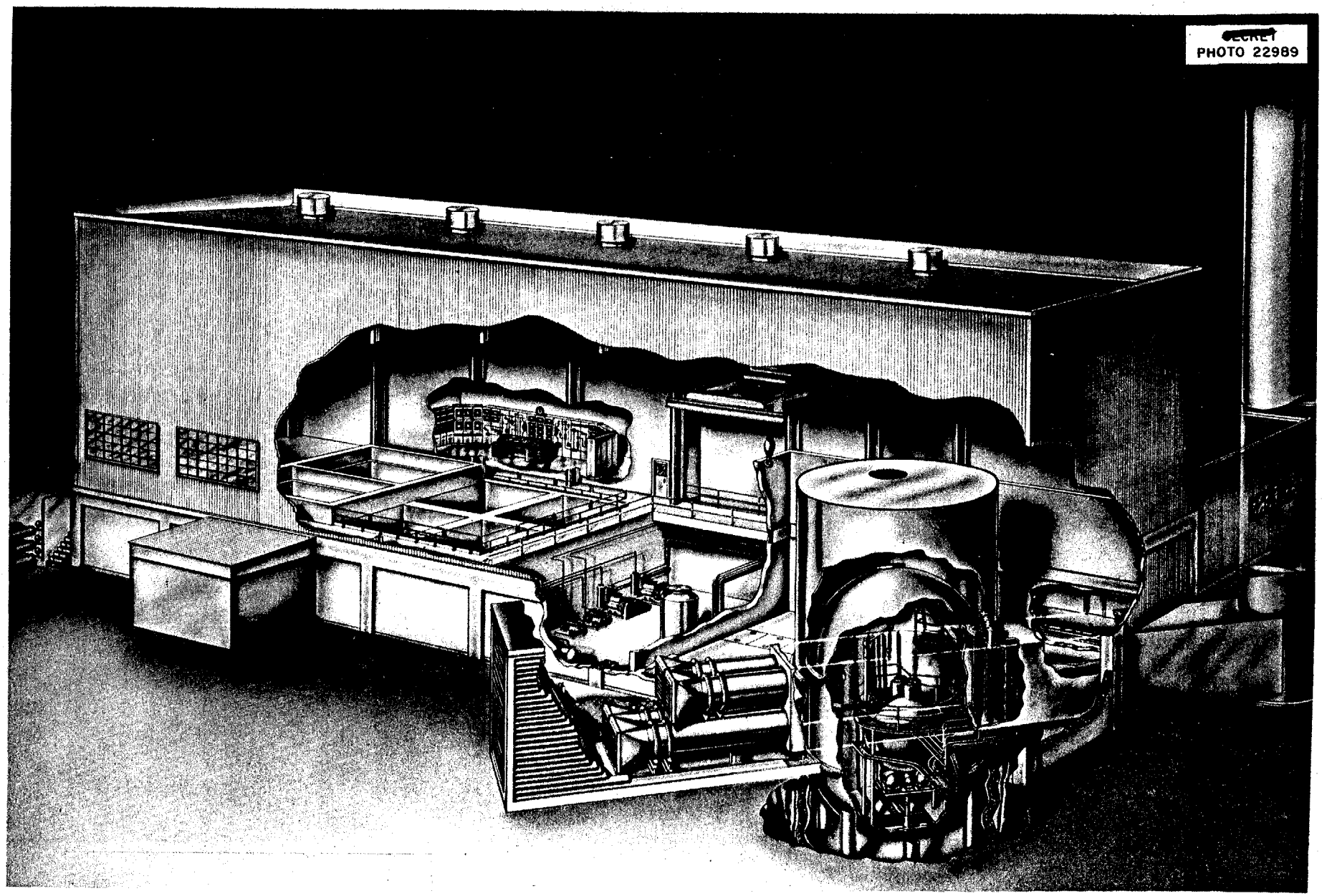

Fig. 3.1. Aircraft Reactor Test Facility.

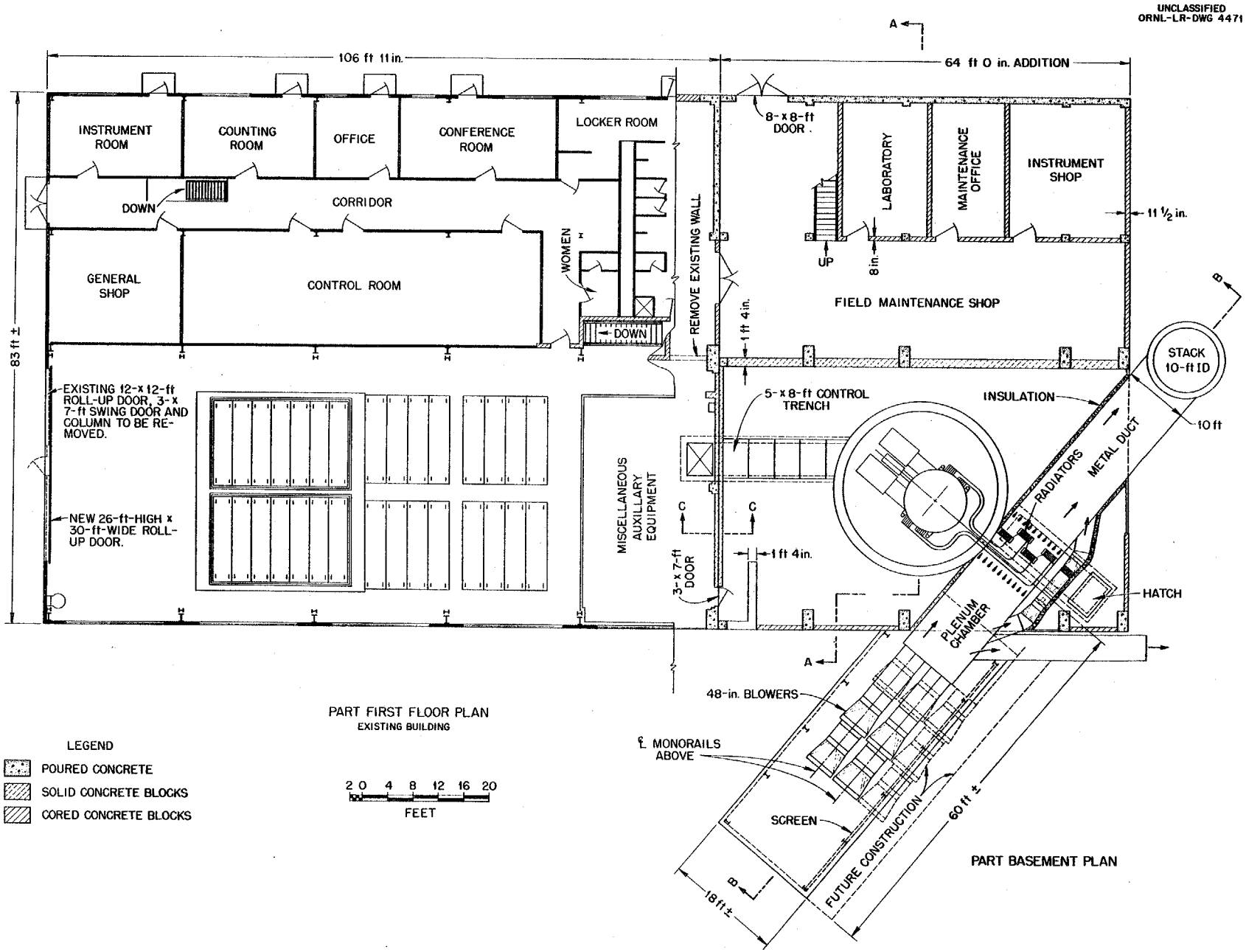

Fig. 3.2. Plan of the ART Building.

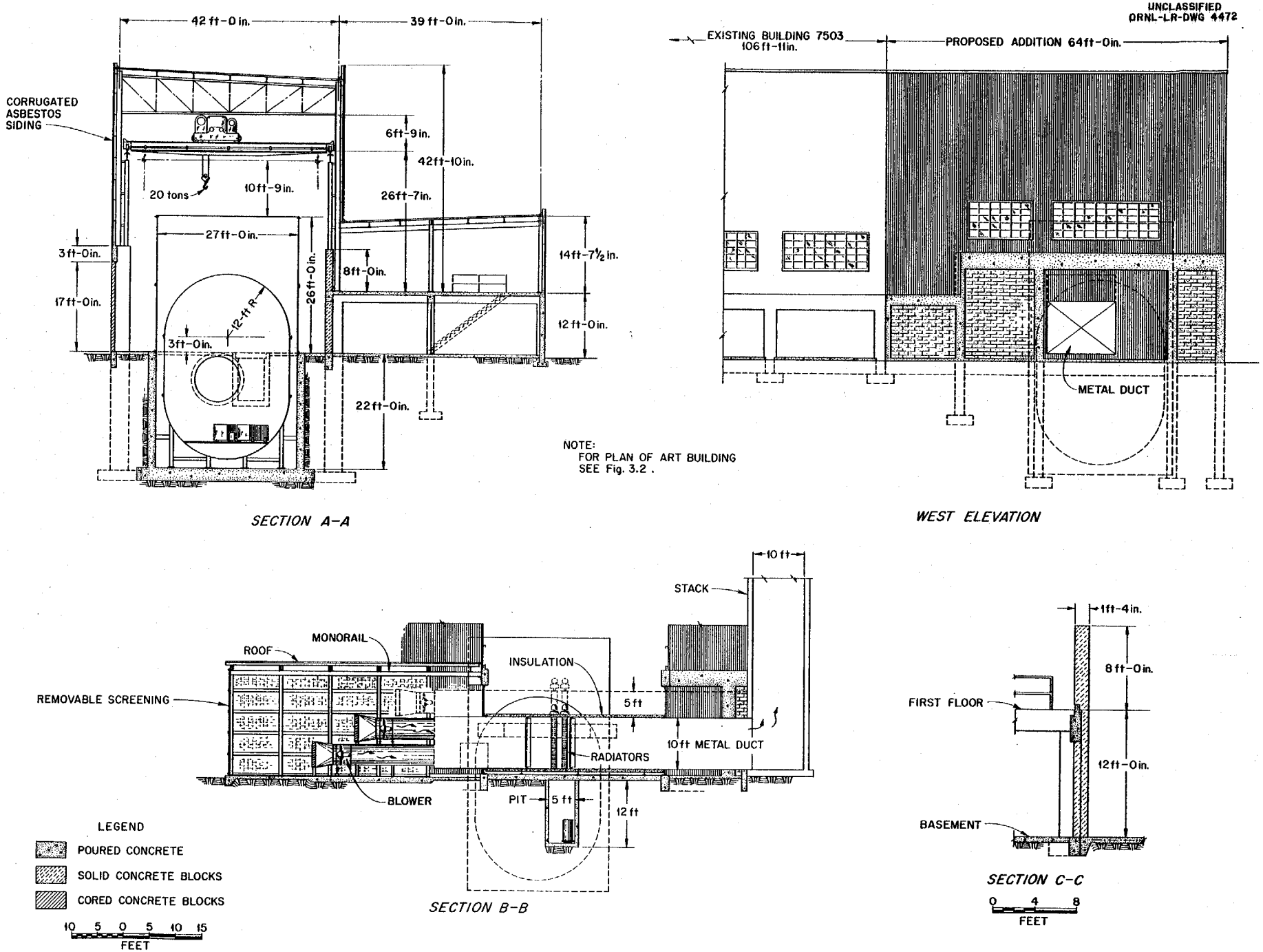

Fig. 3.3. Elevation of ART Building.

and possibly for underwater reactor disassembly work after reactor operation. Also, the truck door in the north wall of the ARE Building will be enlarged to provide a large entry door to the ART area.

Field maintenance and laboratory facilities will be installed in the area east of the new bay and south of the low bay of the ARE. This area and the old experimental bay will be partitioned from the new bay with about a 16-in.-thick solid, concrete-block shield wall. This wall will not be erected until after placement of the upper sections of the reactor assembly container. The only other major modification to the ARE facility to accommodate the ART will be that of modifying and equipping one of the ARE experimental pits for underwater disassembly work on the reactor after operation.

Reactor Cell

The cell designed for housing the reactor assembly is shown in Fig. 3.4. As may be seen, the cell is to consist of an inner and an outer tank. The heat dump equipment will be located outside the cell, but nearby. The space between the two tanks will be of the order of 18 in. and will be filled with water. The inner tank will be sealed so that it can contain the reactor in an inert atmosphere of nitrogen at atmospheric pressure, but it will be built to withstand pressures of 100 psi. The outer tank will be merely a water container.

The inner tank will be approximately 24 ft in diameter with a straight section about 11 ft long and a hemispherical bottom and top. The outer tank, which is to be cylindrical, will be approximately 27 ft in diameter and about 47.5 ft high. When the reactor is to be operated at high power, the space between the tanks and above the inner tank will be filled with water so that in the event of an accident so severe as to cause a meltdown of the reactor the heat given off by the decay gamma activity will be carried off by the water. Since the heat transfer rate to water under boiling conditions would be exceedingly high (of the order of 320,000 Btu/hr·ft^2) and since the thermal conductivity of the fluoride fuel is relatively low, the water-side temperature of the inner tank will not exceed the water temperature by more than 40°F. The water capacity of the space between the tanks together with the water in the reservoir above the inner tank (approximately 10 ft deep) will be of the order of 1,000,000 gal. Boiling of the water in the annulus and above the inner tank will suffice to carry off all the heat generated by the fission products after any accident without any additional water being supplied to the tank.

About 26 ft of the outer tank will be above floor grade. This portion of the tank, as well as the top hemisphere of the inner tank, will not be attached until completion of the reactor installation and preliminary shakedown testing. Since the shielding at the reactor will be quite effective, the space inside the tank will be shielded fairly well so that it will be possible for a man to enter the inner tank through a manhole for inspection or repair work, even if the reactor has been run at moderately high power.

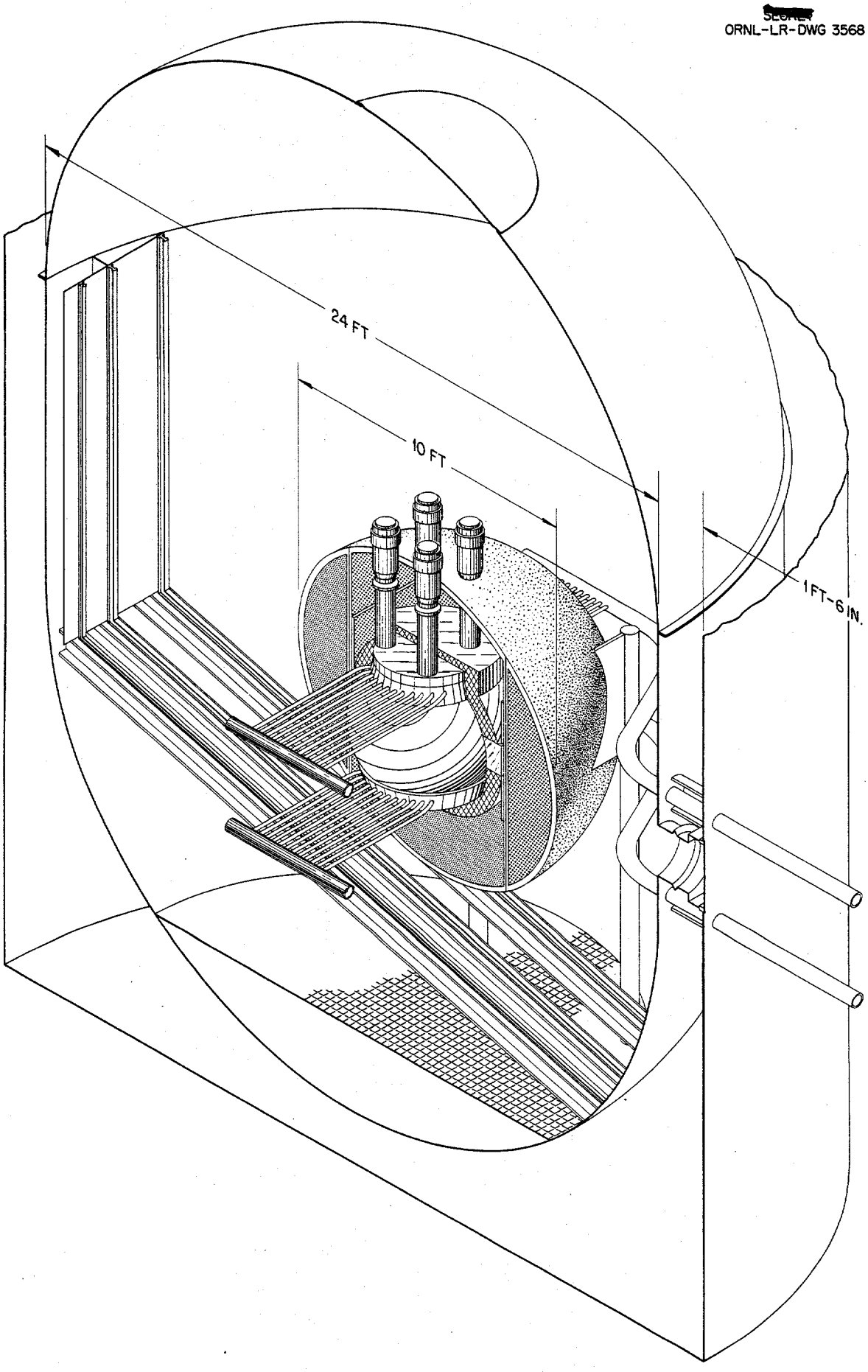

Fig. 3.4. Reactor Assembly Cell with Water-Filled Annulus.

The unshielded reactor assembly will weigh approximately 10,000 lb; the lead gamma shield, approximately 30,000 lb; and the water in the shield, approximately 34,000 lb. The first two of these items can be handled conveniently with a 20-ton crane, while the borated water will be pumped in after the rubber tanks have been installed for the water shield.

The reactor assembly, with its aircraft-type shield, will be mounted in the inner tank on vertical columns with the reactor off-center from the vessel axis and about 6 ft above an open-grated floor. This positioning will provide the space needed for movement of the portable fluoride fuel and sodium moderator coolant containers to their operating stations under the reactor. The off-center location will also serve to minimize the length of the NaK piping.

The NaK and off-gas piping connected to the reactor will pass through a thimble-type passage or bulkhead in the double-walled cell. The openings will be covered with stiff plates which will be welded to the tank walls. The piping will be anchored to the inner plate and connected with a bellows-type seal to the outer one. The volume within all bulkheads will then be maintained at a pressure above that of the inner tank to prevent out-leakage from the inner tank.

A doubly-sealed junction panel for controls, instrumentation, and auxiliary services will be installed through the tank below the building floor grade as a part of another bulkhead to pass wires, pipes, tubes, etc., required for the circuits and systems. The various thermocouples, power wiring, etc., will be installed on the reactor assembly in the shop and fitted with disconnect plugs so that they can be plugged into the panel in a short period of time after the reactor assembly has been lowered into position in the test facility. This will minimize the amount of assembly work required in the field.

Two more bulkheads, in the form of manholes, will be installed in the upper portion of the container. One manhole will be about 3 ft by 5 ft and located just above the flange on the inner tank to allow passage through both container walls and thus provide an entrance to the inner tank for use after placement of the upper sections of the container. The second opening will be a manhole about 5 ft in diameter in the hemispherical top of the inner container to provide overhead crane service after placement of the top. Sufficient catwalks, ladders, and hoisting equipment will be installed within the inner tank to provide easy access for servicing all equipment.

The control bulkhead in the cell will be located so that the associated control junction panel and the control tunnel will extend to the auxiliary equipment pit (formerly the ARE storage pit). The pit and basement equipment will include such items as the lubricating oil pumps and coolers, borated shield water make-up and fill tank with a transfer pump, vacuum pump, relays, switch gear, and emergency power supply. The reactor off-gas flow, diluted with helium, will be piped through the NaK piping bulkhead to the disposal facility outside the building.

Heat Dump System

The basic requirement of the ART heat dump system is to provide heat dump capacity equivalent to 60 Mw of heat with a mean temperature level of 1300°F in the NaK system. It has also seemed desirable that the ART heat dump system should simulate the turbojet engines of the full-scale aircraft in a number of important respects, such as thermal inertia, NaK holdup, and fabricational methods.

Since heat dumps are required for use in heat exchanger test rigs, and since work of this character has already progressed to the point where the cheapest, most compact, and most convenient heat dump is currently proving to be a round-tube and plate-fin radiator core, it is believed that this basic type of heat transfer surface should prove both sufficiently reliable and sufficiently well-tested to serve for the ART. The round-tube and plate-fin radiator planned for the ART makes use of type 310 stainless steel clad copper fins spaced 15 per inch and mounted on 3/16-in.-OD tubes placed on 2/3-in. square centers. Individual radiator cores will have an inlet face 2 ft square. Much information has been obtained in tests of similar units.[2]

NaK will be circulated through five separate systems. Four will constitute the main heat dump system, while the fifth will be the moderator heat dump system. In the main heat dump system a group of four fill-and-drain tanks will be used, but these will not require the remotely operable couplings desired for the sodium or fuel systems. The NaK will be forced into the main cooling circuit by pressurizing the tanks. The 24 tube bundles of the fuel-to-NaK heat exchanger will be manifolded in four groups of six each. The NaK will flow from these tube bundles out to the radiators which will be arranged in four vertical banks with four radiator cores in each bank. The NaK will flow upward through the radiator bank to the pumps. A small by-pass flow through the expansion tank will allow it to serve as a cold trap. A filter to remove oxides will be placed in the return line from the tank.

The moderator heat dump system will be essentially similar except that its capacity will be about one-quarter that of one of the four circuits of the main heat dump system. NaK will be circulated to the Na-to-NaK heat exchanger in the top of the reactor where the NaK will pick up from the sodium the heat generated in the island and reflector. The NaK will pass to a small NaK-to-air radiator where it will be cooled and returned to the pump suction. An expansion tank and by-pass filter will be included, as in the main NaK system. It is planned to have only drain and filter by-pass throttle valves in the NaK systems, since the NaK will be drained if any repairs are required.

As shown in Fig. 3.2, the NaK-to-air radiators will be mounted in an air duct close to the reactor cell. This duct traverses the southwest corner of the building addition. The radiators will be located at floor grade

2. W. S. Farmer et al., Preliminary Design and Performance of Sodium-to-Air Radiators, ORNL-1509 (Aug. 26, 1953).

over the NaK pipe-line pit. Four axial-flow blowers will force 300,000 cfm of air through the radiators and out through a 10-ft-dia discharge stack 78 ft high. Since the axial-flow blowers will stall and surge if throttled, control can be best accomplished through by-passing a portion of the air around the radiators. This arrangement requires only constant speed a-c motors and simple duct work with a controllable louvre for by-passing. The heat dump rate will be modulated by varying the number of blowers in operation. A set of counterweighted self-opening louvre vanes in the inlet or discharge duct from each blower will prevent backflow through the blowers not in operation. Thus each blower will be driven with an a-c motor independently of the others, and the heat dump capacity can be increased in increments of 25% from zero to full load. An additional 4 ft by 8 ft set of controllable louvres will be mounted in such a way as to bleed air from the plenum chamber between the blowers and the radiators to get vernier control of the heat load.

Heat barriers mounted on either side of the radiators will be required to minimize heat losses during the warmup operations. Warmup will be accomplished by energizing the pumps and driving them at part or full speed. Since approximately 400 hp must be put into the pumps in the NaK circuits, this power will appear as heat in the fluid pumped as a result of fluid frictional losses. A mechanical power input of 400 hp to the NaK pumps will produce a heat input in the NaK system of approximately 300 kw. This should be enough to heat the system quite satisfactorily with the radiator cores blanketed to prevent excessive heat losses. Relatively simple sheet stainless steel doors filled with 0.5 in. of thermal insulation when closed over both faces of a radiator (64-ft^2 inlet-face area) filled with 1100°F NaK will give a heat loss of 30 kw.

Heat load control for the low-power range presents some problems. The plenum chamber pressure with the 4 ft by 8 ft by-pass louvres wide open and one blower on will be about 0.5 in. H_2O. This will give a heat dump capacity of 3 Mw if all the radiator heat barriers are opened. Lower heat loads can be obtained by varying the number of heat barriers opened. Operating the heat barrier doors against a pressure of 0.5 in. H_2O (2.6 lb/ft^2) should not be difficult.

The heat appearing in the moderator will be about 3.5% of the reactor power output. The moderator cooling circuit will also remove heat from the core shells and the pressure shell so that the total amount of heat to be removed from the moderator cooling circuit will be about 6% of the reactor output. This must be removed at a mean NaK circuit temperature of about 1050°F. A radiator having a 2-ft^2 inlet-face area and the same proportions as those used for the main heat dumps will be employed. This radiator will be supplied by a 2-ft-dia blower operating at about 3400 rpm.

Fill-and-Drain System

To permit fairly easy fuel loading and removal, an effort is being made to develop a good, reliable, relatively simple fill-and-drain system incorporating a remotely operable coupling. Such a piece of equipment will permit

removal of the fuel from the reactor and provide considerable flexibility in the conduct of operations. It is believed that by removing the fuel the radiation level can be cut for maintenance operations. Easy installation or removal of the fuel will also facilitate reprocessing the fuel or modifying the fuel composition. Should a reliable remotely operable coupling not be developed in time, welded attachments will be made between the various fill tanks and the fluid systems.

For handling the heavy, shielded fluoride and sodium containers inside the pressure vessel, a track will be installed on the floor and inside the wall. Wheels will be mounted on the tank dolly, both on the bottom and on one end, so that the assembly can be lowered by the overhead crane to the floor track with the end wheels on the dolly rolling against the vertical track. Once on the floor track, each dolly will be moved to its operating station under the reactor. Each track pair in this area will be mounted on a lift for raising the tank connection nozzle to the contact position within the reactor shield.

Other requirements of the fill-and-drain system include provision for accurate measurement of the quantity of fluid in the drain tank at all times during either the filling or the draining operation. This is particularly important in connection with reactor fuel systems because it is important that the exact amount of fuel in the reactor be known at all times.

The shielding required for the fuel tank will be 10 in. of lead to reduce the dose to 1 r/hr at 5 ft from the tank one week after full-power operation. The resulting shield weight for a 7-ft^3-capacity tank will be about 15 tons. The lead shield required for the 1-ft^3 sodium drain tank will be 5 in. thick, and it will weigh approximately 2 tons.

Fill and drain systems for fuel or for sodium from the moderator circuit will include provision for both preheat and the removal of decay gamma heat. These functions will be carried out by diverting NaK from the radiator circuits and directing it through a jacket surrounding the drain pipe and through coils in the drain tank.

Off-Gas Disposal System

The design of the off-gas system was based on the pessimistic assumption that all the fission products will be given up to the off-gas system as they are formed and will be swept out with 1000 liters of helium per day. They will be passed through a long charcoal-filled pipe designed so that no more than 0.001 curie/sec of radioactivity will go up the stack. About 4 Mw of the 60 Mw will appear as fission-product decay energy. Since the design was based on all of the fission products being released to the off-gas system, a decay of $t^{-0.2}$ was used in calculating the heat released.

The gases will be removed from the fuel in the expansion tank at the top of the reactor and vented through a 1/4 in. Inconel line 20 ft long to a 2-in. steel pipe 1050 ft long. The first 50 ft of this pipe will be open, but the balance will be filled with activated charcoal. The gas

will exhaust from the charcoal bed to the stack where it will mix with the 670,000 cfm of hot air from the radiators. On this basis, the holdup time and heat generation for each component of the off-gas system will be as follows:

Component	Volume (cm^3)	Holdup Time (sec)	Heat Generation (kw)
Expansion tank (gas volume)	7×10^3	600	3000
1/4 in. line 20 ft long	200	17	10
2 in. line 50 ft long	3.4×10^4	3×10^3	325
2 in. line 1000 ft long (assuming 1/2 volume is charcoal)	3.4×10^5	3×10^4	665

The entire 1050 ft of 2-in. pipe will be in a trench under 6 ft of water for heat dissipation. Additional shielding will be used as needed.

If the gases are held up for one or two days, calculations show that Kr^{88} presents the greatest hazard, since Kr^{88} and its daughter Rb^{88} give out about 2.4 Mev per disintegration of Kr^{88}. The activity in curies/sec is

$$\frac{6 \times 10^7 (w) \times 3 \times 10^{10} \text{ (fissions/sec·w)} \times 0.04 \text{ (atoms/fission)} \times \lambda e^{-\lambda t}}{3.7 \times 10^{10} \text{(dis/sec·curie)}}$$

$$= 1.9 \times 10^6 \lambda e^{-\lambda t} .$$

For a decay constant of 6.9×10^{-5} dis/sec·atom and after a holdup time of 48 hr, there will remain less than 10^{-3} curies/sec.

The amount of activated charcoal required to assure holdup of krypton may be evaluated as follows:[3]

$$t = \frac{KLA}{f}$$

where t is the holdup time, LA is the length times area (volume) of the charcoal bed, f is the volume flow rate of the off-gas, and K is a constant with a magnitude of about 500 for the type of charcoal to be used. For a holdup time of 48 hr and a flow rate of 1000 liters/day, it is found that only 4 liters of charcoal will be needed, whereas 600 liters will be available.

3. The formula, which was obtained from M. T. Robinson of the ORNL Solid State Division, is based on experimental evidence.

Since about 0.5 Mw of heat will be given up by the fission products on the charcoal, the limiting factor will be the heat transfer. Upon entering the charcoal, the gas will be absorbed very rapidly, and the first few feet of charcoal will soon rise in temperature to around 400°F. The temperature will start to decrease in less than 10 ft, and, by the time the gas has passed through about 100 ft of charcoal, the temperature will be down to near the ambient temperature of the surrounding water.

A by-pass line around the off-gas system is to be provided for use in case there is a leak along the pipe in the trench. In such an event, the reactor will be shut down and the gas will be vented into the reactor cell. After a two-day holdup, a vent line will be opened directly to the stack. This auxiliary vent line can also be used if fission-product gases leak from any of the reactor components into the reactor cell. Monitrons will be provided at suitable locations in all gas lines. In the event that the off-gas system is to be operated at a time when no power is being abstracted from the reactor, the air from the blowers will be ducted around the radiators to avoid difficulties which would otherwise follow from cooling of the NaK.

4. CONTROLS AND OPERATION

Control Philosophy

The early ORNL effort to develop the circulating-fuel type of aircraft reactor was motivated in part by a desirable control feature of such reactors. This feature is the inherent stability at design point of the over-all power plant that results from the negative fuel temperature coefficient of reactivity. In a power plant with this characteristic the nuclear power source is a slave to the turbojet load with a minimum of external control devices.

This predicted master-slave relationship between the load and power source was verified by the ARE. Controlwise the power plant consists of the nuclear source, the heat dump (in the case of ART), and the coupling between source and sink (the NaK circuit). Control at design point can be effected to some extent by nuclear means at the reactor, by changing the coupling (i.e., changing the NaK flow), or by changing the load (i.e., the heat dump from the NaK radiators).

For the ART at design point the regulating rod will be used mainly for adjusting the reactor mean fuel temperature. In particular, an upper temperature limit will cause the regulating rod to insert until the fuel outlet temperature does not exceed 1600°F. This limit will override any normal demand for rod withdrawal. Furthermore a low NaK outlet temperature from the heat dump radiators will automatically decrease the heat load to keep the lowest NaK temperature of the system at no less than 1050°F. This lower temperature limit will override all other demands for power.

Critical experiments will be performed with the system isothermal at 1200°F. This temperature was chosen to increase the life expectancy of the beryllium moderator. The mean fuel temperature at design point will be 1400°F, the moderator being held at 1200°F. Since the fuel temperature coefficient is 5×10^{-5} per °F, 1.0% $\Delta k/k$ will be required to raise the mean fuel temperature this 200°F.

The total worth of the regulating rod over its stroke will be about 5% $\Delta k/k$. In addition to its use in changing the mean fuel temperature, this amount of rod will supplement the fuel addition by solid pills (discussed at the end of this chapter) in compensating for burnup and fission-product poisoning.

For operation in the design-point range (from 20 to 120% of design-point power, which is the useful range for an aircraft power plant), a manual change in load demand or in operating temperature will be restricted by the maximum rate at which the load can be changed or by the maximum rate at which the regulating rod can be withdrawn, respectively. In the design point power range the maximum rate of withdrawal of the regulating rod will be obtained by adding $\Delta k/k$ at the rate of 3.33×10^{-4} per second.

This will raise the fuel outlet temperature at the rate of 12°F per second until the maximum fuel outlet temperature of 1600°F is reached, at which point the temperature limit will hold. In this range a permissible load change rate of one-half design point power in 1 min is comparable to the requirements of engine performance for a nuclear-powered aircraft. Load changes are effected by manual demand for changing the air flow over the NaK radiators.

Control of the ART is classified in three different categories of operation: namely, (1) startup, (2) operation between startup and design point, and (3) operation in the design point range. For the second and third of these categories the nature of the reactor and power plant is so different from that of conventional high flux reactors that control must be based on inherent characteristics of the reactor to a large extent rather than on conventional reactor control art. There is no conventional art for these categories with high flux reactors. Control at startup utilizes, in principle, old reactor control art with short-period "scrams" that are conventional in principle. Experimentation will take place primarily in the startup and design-point regions. In the intermediate region between these two, little testing will take place. Consequently, operational procedure will be followed to take the reactor from the low-level adequately controlled region to the high level region in one simple manner. This procedure will be assured by permissive instrument interlocks that are described in a following section of this chapter.

Fission chambers and compensated ion chambers will be located beneath the reactor shell between the fill and dump tanks and the reactor. The region around the pipes between these tanks and the reactor will be filled with moderator material, either Be or BeO, through which cylindrical holes for these chambers will run radially out from the centerline of the system. From four to six such holes will be available. Chamber sensitivities will be adequate for the entire range of nuclear operation.

The fuel expansion chamber is a key item in providing safety for the ART. The fuel temperature coefficient of reactivity provides stability for the system by the expansion of fuel from the critical region. Adequate expansion volume will be available at all times.

Scram System

A conventional scram system achieved by dropping poison rods into the critical lattice will not be used with the ART. The reasons for eliminating this feature are the following:

1. In the design-point range described, analysis shows that limiting the rate of rod withdrawal and the rate of load increase will limit the period of the reactor when it is operating normally. A limited rate of rod withdrawal and a limited rate of load increase near the design point in the ARE gave a minimum period of about 10 sec. The same technique will be used on the ART.

Short periods, of the order of 1 sec, can occur in the design-point range only in the event of structural failure. The total $\Delta k/k$ required in the rod to override an increase resulting from such failure cannot be obtained from one rod nor could such a rod, were it available, be inserted fast enough to prevent a serious accident. The temperature coefficient will react so rapidly that it will limit the signal which would normally actuate a scram, except for an extremely high rate of increase in reactivity. It has not been possible to devise a control system that would react rapidly enough in such cases to prevent the accident. Therefore, in the design-point region the conventional scram would be of little merit.

2. For the initial loading and critical experiments a scram system will be used, but it will not involve dropping the one control rod. The method described below was proposed because the single-rod system lacks the safety feature of a plurality of rods, as ordinarily found in conventional reactors. The actuating signal will be a short period, a high flux, a manual scram, or any of a number of failures in the system, and the signal will be supplied through an auction circuit in the conventional manner.

Essentially, the safety of the system lies in the procedure of adding fuel in a subcritical external loading tank and forcing it against gravity into the fuel system. This will be done by pressurizing the loading tank with helium through a valve which will fail closed. Two parallel helium outlet lines from the loading tank to the off-gas system will fail open. All zero power tests and measurements will be made with the valve between the loading tank and the reactor locked open, and the signal from the auction circuit will actuate the solenoid in the helium-pressurizing system in the manner described above. Actually, two parallel dump lines, one to the fill-and-drain tank and one to the emergency dump tank, will contain valves which will be actuated simultaneously on the dump signal from the auction amplifier circuit. This system has the merit that if too rapid addition of fuel to the system causes a short period, reversal of the operation will reverse the period. Control of the helium pressurizing system will limit the rate at which the fuel is added to the system. The helium system can be designed so that it will fail safe, except for the case of a plurality of simultaneous failures comparable in probability to the failure of a plurality of magnetic clutches all of which simultaneously fail to open in the conventional rod-dropping reactor scram.

Startup

A rather close estimate of the critical concentration should be available from the hot critical experiment so that 80% of the uranium will be in the fuel at the time of the ART startup. The final 20% of the required uranium will be added in steps. After each uranium addition the fuel will be forced from the fill tank up into the reactor by means of helium pressure. The scram circuit will be available, as described previously. The rod will be inserted for each step, a given amount of uranium will be added, the rod will be slowly withdrawn, and a count will be taken on the fission chambers to determine the subcritical multiplication as a function of uranium concentration. A polonium-beryllium source of approximately 15 curies strength will be installed in the central island of the reactor to provide neutrons for startup.

Before beginning the critical experiment, the speed of the pumps will be set so that the flow through the core will be about 50 gpm. Thus over one-half the delayed neutrons will be available for control while going critical. The whole system is to be isothermal at 1200°F.

Control for zero power operation (rod calibration, fuel enrichment, determination of the temperature coefficient, etc.) will be manual with the maximum rod speed providing a rate of change in $\Delta k/k$ of 3.33×10^{-4} per second. Overriding the manual rod withdrawal will be a 5-sec period rod reverse and a 1-sec period fuel dump by relieving the helium pressure in the loading tank, as described above.

A holding servo system will be used at zero power for experiments requiring constant neutron flux. Operation with the servo system will be essentially the same as that for the ARE. Limits will be maintained on the rod speed, and overrides will be maintained on period and temperature.

Operation Between Startup and Design Point

Since most of the nuclear data will have been obtained from the hot critical experiment, the low power operation will be held to a minimum. After going critical, the reactor will be leveled out manually at about 10 to 100 watts. The pumps will be stopped, and the reactor will be allowed to go on a period. This will be a check on the effects of flow rate on the reactivity contribution of the delayed neutron fraction. When the power level has reached about 1 kw, the pumps will be started and the rod will be inserted to drive the reactor subcritical. Sufficient uranium will be added to give about 0.5% excess reactivity. The pumps will again be stopped and the reactor will be brought to about 10 watts; the rod will be withdrawn; and the reactor will be allowed to go on a period until a level of 1 kw is reached. The pumps will again be started, and the rod will be inserted to drive the reactor subcritical. This procedure will be repeated to give 2 or 3 calibration points on the regulating rod. Since a similar rod will have been carefully calibrated in the hot critical experiment, only a few rough check points will be necessary.

The power level will then be elevated to about 10 kw and leveled out manually with the rod. At this power level, the shielding and off-gas systems will be checked out thoroughly without great hazard to personnel. The pump speed will then be increased so that the fuel flow rate will be increased from 50 gpm to the design flow rate of 1200 gpm. This will cause a decrease in reactivity of the order to 0.2% Δk, and the rod will be withdrawn accordingly. The reactor will then be ready to deliver power.

The negative fuel temperature coefficient of the ART makes manual control mandatory in taking the reactor from zero power to some power at which the temperature coefficient provides stability while the reactor gets its power demand from the load. Accordingly, a single operation procedure for every operation in this range will be followed. The load will be interlocked so that permission to start adding the load will come only when a compensated ion chamber current reaches some prescribed value. This value will be determined in the manner described below the first time the reactor is taken to power.

With all loop flow rates at design point and the reactor at about 10 kw and isothermal at 1200°F, the regulating rod will be withdrawn until the reactor is on a positive period. This period will gradually increase until it becomes infinite and finally negative because of the temperature coefficient. Meanwhile, both the log N and micromicroammeter readings will go through a maximum. This maximum log N reading will provide the signal to permit opening of the heat barrier doors to the NaK radiators. Natural convection from the radiators with these doors open will be about 300 kw. Accordingly if these doors are opened when the log N reading reaches this value, the temperature coefficient will always suffice to provide regulation and stability, provided the rate of load demand above this minimum is restricted to the values cited previously.

If after shutdown the flux exceeds the log N reading required to open the heat barriers, opening of the barriers will not "shock" the system even though the reactor may be subcritical at the time the doors are opened. If on the other hand the flux is too low to permit opening of the barriers, there is only one procedure for getting permission.

Design-Point Operation

With the reactor at about 300 kw (estimated from the power extracted by opening the heat barrier doors) the blowers will be started and heat will be extracted from the NaK, which, in turn, will extract heat from the fuel. The reactor will be leveled out at 3, 15, 30, and 60 Mw. A heat balance will be obtained at each level of extracted power vs nuclear power. The operation of all components will be observed at each power level. Care will be taken not to exceed the maximum temperature of 1600°F or fall below the minimum temperature of 1150°F. The reactor will then be operated for 1000 hr at 60 Mw.

Xenon will be removed continuously from the fuel by helium injected into the pump chamber and escaping in the swirl chamber. The rate of removal by this means can be determined only by operating the power plant. However, experience with the ARE has indicated that less than 1% $\Delta k/k$ of the regulating rod will be needed to cope with the xenon that is not removed. Should the purging be much less than is anticipated, the xenon could and would under some circumstances shut the reactor down. The low-temperature limit on the NaK radiator outlet temperature will automatically remove the load to effect this shutdown. In case this happens the fuel will be dumped until the Xenon decays.

Fuel enrichment to compensate for burnup will be accomplished by adding fuel in the form of high-U^{235}-content "pills" of solid fluoride fuel. These will be introduced into the reactor fuel circuit through an entry provided in the fuel expansion tank located on the north head of the reactor. The pill-addition mechanism will be carefully designed and tested to make it jam-proof and incapable of ejecting all its pills in one spurt. It will permit the introduction of only one pill at a time to the reactor system.

The total burnup is equivalent to about 2.5% $\Delta k/k$. The capacity of the pill machine will be such as to hold no more pills than that amount equivalent to 2.5% $\Delta k/k$. Accordingly, the rod will always be capable of overriding any fuel addition. Furthermore, the rate at which successive pills can be added will be much less than that which can be effectively cancelled by movement of the control rod. Compensation for burnup and fission-product poisoning can be accomplished by both control rod withdrawal for fine control and by fuel enrichment for coarse control.

The best choice for a pill material is the compound Na_2UF_6. This was used as the enriched fuel component for the ARE test. Its melting point is approximately 1160°F, and its solid density at 1000°F is about 4.9 g/cm^3. It is composed of approximately 60% U^{235}, 11% Na, and 29% F (by weight). Several pill dispenser (and container) designs have been prepared that are based on the use of pills 1/2 in. in diameter by 1/4 in. thick. Pills of this size have a volume of 0.80 cm^3, they weigh 4.0 g, and they contain about 2.4 g of U^{235}. The rate of pill addition required to maintain a constant reactor fuel inventory will thus be 0.5 pill per Mwd of operation. For operation at a power level of 60 Mw, this will require the addition of 30 pills per day, or a total of about 1200 pills for the 1000 hr of full-power operation.

5. REACTOR HAZARDS

An attempt has been made to envision as many hazards as possible that might occur during the course of the operation of the Aircraft Reactor Test. Included in this chapter, therefore, is a discussion of the normal radiation hazards, the hazards resulting from operational or equipment failures, and fluid leaks, as well as the nuclear and chemical hazards peculiar to the cycle. The dispersion of airborne activity, either from the off-gas system or following a hypothetical accident in which all the fuel is volatilized, is described in the following chapter, "Dispersion of Airborne Activity."

The radioactivity of the ART will be inherently confined by the nature of the design and materials in such a manner that the uncontrolled dispersion of the activity outside the reactor cell will be virtually impossible. Consequently, the hazard from most failures will be negligible, since the only action required will be dumping of the fuel; the activity would not even be released to the cell. Furthermore it is shown that while a hypothetical nuclear accident could rupture the reactor pressure shell, the reactor cell would remain intact and the accident would be safely contained.

Some consideration has been given to cases in which the reactor cell, as well as the pressure shell, would be ruptured, and the resulting subsequent dispersion of activity has been examined in detail. It is believed that such an accident could occur only as the result of aerial bombing or sabotage.

Radiation Dose Levels

The radiation dose levels to be expected at representative stations at the facility have been estimated for a variety of conditions and have been tabulated in Table 5.1. The shielding assumed for these estimates was the following: (1) the primary aircraft-type reactor shield designed to give 1 rem/hr at 50 ft at full power, (2) the reactor cell steel and water walls, (3) 16 in. of concrete block stacked around the reactor cell to a height of 10 ft, (4) 16. in of concrete block stacked between the reactor room and the maintenance shop and between the reactor room and the former ARE main test bay, (5) concrete block stacked around and on top of the air duct for the NaK-to-air radiators in such a way that the equivalent of 12 in. of concrete will be imposed along any radial line extending outward from the radiators.

The concrete blocks to be stacked around the periphery of the reactor test room and around the air duct and radiators are intended to provide shielding in case a fuel leak developed either into the reactor cell or into the NaK systems.

TABLE 5.1. RADIATION DOSE LEVELS

Location	NORMAL OPERATION DOSE LEVEL (rem/hr)			DOSE LEVEL (rem/hr) WITH 1% OF FUEL IN NaK IN RADIATOR			DOSE LEVEL (rem/hr) WITH ALL FUEL IN BOTTOM OF CELL	
	Full Power	15 min After Shutdown	10 days After Shutdown	Full Power	15 min After Shutdown	10 days After Shutdown	15 min After Shutdown	10 days After Shutdown
Reactor shield surface	100	4	1	3×10^5	10^5	2×10^4	10^4	2×10^3
Outside reactor cell	10^{-2}	4×10^{-4}	10^{-4}	1500	500	120	60	15
Outside reactor room	10^{-4}	4×10^{-6}	10^{-6}	3	1	0.25	0.3	0.08
Control room	10^{-5}	4×10^{-7}	10^{-7}	0.6	0.2	0.05	0.04	0.01
Road	5×10^{-5}	2×10^{-6}	5×10^{-7}	0.6	0.2	0.05	0.1	0.02

As may be seen from Table 5.1, the reactor will be adequately shielded so that the control room operators will receive much less than 1 rem/hr even with 1% of the fuel in the radiator and much less than 0.1 rem/hr even if the pressure shell is ruptured. The dose rates would be considerably higher, however, if it were postulated that the reactor cell was also ruptured, in which case the activity would no longer be confined. This extreme situation is considered in the following chapter, "Dispersion of Airborne Activity."

Typical Operational and Equipment Failures

There are any number of operational or equipment failures that can be envisioned in a system as complex as the Aircraft Reactor Test. In this section are listed those failures which would have the greatest effect on the operation and which therefore seem to offer the greatest hazards. For each failure some probable causes are given, as well as the result, and the action required in order to minimize the hazard is stated. In all cases, as will be shown, the failure would be inconvenient, but no serious danger would ensue, since the most drastic action required would be dumping of the fuel (and/or sodium) into the dump tanks. Therefore it is also apparent that the operability of the dump system must be assured.

Fuel Freeze. Excessive cooling of the primary NaK circuit could cause fuel to freeze in the heat exchanger and stop the flow of the fuel. The heat exchanger would not be seriously damaged, but some cracks might form in tube walls. Because of fuel flow stoppage and consequent lack of cooling, the temperature of the fuel in the fuel circuit would rise 13°F/sec (boil in 2 min) as a result of fission-fragment decay heat, which will be 3800 Btu/sec (6% of power) immediately upon cessation of cooling in the circuit. In the reactor structure, the cooling available from conduction after dumping of the fuel would not be adequate to keep pump blades, wells, and other points where fuel might be trapped from being raised to fuel vaporization temperature. Excessive cooling in the primary NaK circuit would, therefore, require that the fuel be dumped.

Sodium Freeze. Excessive cooling of the secondary NaK circuit could cause sodium to freeze in a sodium-to-NaK heat exchanger and stop flow in the moderator cooling circuit. The temperature of the beryllium moderator would increase 0.5°F/sec at full-power operation and 1°F as a result of decay heat of activated materials in the moderator region. It would be necessary to dump the fuel immediately.

Structural Failures. Corrosion, excessive heating, and pressure surges would be the possible causes of structural failures. If a pressure surge caused a 0.020-in. expansion of the outer core shell, there would be a reactivity change of +0.002. If the outer shell were to collapse under an excessive external pressure load, there would be a large reactivity decrease and the possibility of leakage of sodium into the fuel circuit. The results of deformation would be similar, but the effects would be of a lower magnitude. A failure of the pressure shell would release fission products; hot, highly radioactive fuel and attendant decay heat; and NaK.

A failure in the fuel-to-NaK heat exchanger would cause a NaK or a fuel leak. Deformation in the fuel-to-NaK heat exchanger would possibly result in slight changes in pressure drops and heat transfer characteristics. It would probably be necessary to dump the fuel if any of these postulated events occurred, with the possible exception of slight deformation in the fuel-to-NaK heat exchanger.

Pump Failures. Loss of power to pump drives, as well as seizing of shafts, bearings, or impellers, could cause pump stoppage. If one of the two fuel pumps stopped pumping, the fuel flow pattern would be altered and roughly one-half the fuel-to-NaK heat exchanger would be starved. There would be a consequent reduction in power output. If both the fuel pumps failed, fuel flow would stop and the fuel temperature would rise 13°F/sec because of fission fragment decay heat (see item on "Fuel Freeze" above). If only one fuel pump failed the fuel would be dumped or the reactor would be operated at reduced power; if both pumps failed it would be necessary to dump the fuel.

If one sodium pump failed at full power operation, the temperature of the sodium would rise 0.25°F/sec to accommodate the increased heat load on the operable pump, and it would be necessary to dump the fuel or to operate the reactor at reduced power. If both sodium pumps failed, the temperatures of the beryllium in the moderator and the sodium would rise 0.5°F/sec, and it would be necessary to dump the fuel.

The failure of one NaK pump in the primary heat exchange circuit would reduce NaK flow and consequently reduce the reactor power output. As with other pump failures, it would be necessary to dump the fuel or to operate the reactor at reduced power. If all the NaK pumps in the primary heat exchange system failed, the fuel temperature would rise 13°F/sec (see item on "Fuel Freeze" above), and the fuel would be dumped immediately. Failure of the NaK pump in the moderator cooling circuit would cause temperatures of the sodium and the beryllium to rise 0.5°F/sec. As in the case of the sodium pump failures, it would be necessary to dump the fuel.

If the pumps for providing cooling oil to the pumps were to fail, there would be a slow increase in temperature of the oil coolant, the fuel pump shaft, the bearings, and the gas-seal mechanism. Failures of this type would be taken care of by switching to the auxiliary pump and repairing the pump that failed.

Electrical Power Failure. An emergency power supply will be available and all instrumentation and essential equipment would be transferred to it. Therefore, there would be no immediate hazard following such a failure. The emergency power system will be adequate to operate at least one fuel pump, one sodium pump, one NaK pump, one blower, and all the necessary instruments. This equipment will be sufficient to prevent excessive temperature rises from the fuel afterheat. All possible measures will immediately be taken to restore the normal power supply as rapidly as possible. However, if the failure lasts an extended period of time, it may be necessary to dump the fuel.

Fuel Channel Hot Spots. Flow separation in the core or failure of the core-shell coolant system could cause hot spots in the fuel channel. In this event there would be the possibility of fuel boiling in the core and causing irregularities in power or increased corrosion. The power level would be reduced until the fuel boiling ceased, or, if necessary, the fuel would be dumped.

Excessive Fuel Feed. A failure in the enrichment system might result in the addition of excess fuel. In this event the reactor would heat to a new and higher equilibrium temperature. An excess of 0.6 lb of U^{235} introduced instantaneously would make the reactor prompt critical. A $\Delta k/k$ of 0.002 would occur and result in an immediate fuel temperature rise of 40°F. The reactor would quickly level out at the new temperature. If the equilibrium temperature were excessive, the fuel would be dumped.

Fuel Fill-and-Drain System Failure. The fuel fill-and-drain system might fail because of jammed valves or a coolant system failure. If such a failure occurred before the fuel was enriched, there would be no hazard. The system would be repaired, if possible, or the nonradioactive fuel would be drained on the floor of the reactor cell. In the event of an emergency drain of radioactive fuel coincident with a failure of the fill-and-drain system that prevents drainage, the reactor will fail at the weakest point and release hot fuel, fission products, sodium, and NaK in the reactor cell. If the drain system functioned satisfactorily but the dump tank cooling system failed after the fuel was drained, the tank would fail at its weakest point and release large quantities of hot fuel and fission products to the reactor cell. The reactor cell is designed to contain the hot, radioactive fuel, as described in a following section. If it were desired to drain the radioactive fuel under normal operating conditions and drainage was prevented, it would be necessary to cool the radioactive fuel with the normal heat removal system until the decay heat had dropped sufficiently to permit shutdown of the NaK system.

NaK Circuit Heat Dump Blower Failure. If the blowers failed, heat loss from the radiators would be by natural convection only and would be about .3 Mw. Fissioning would stop in the reactor, but the fuel temperature would rise 10°F/sec because of decay heat. In this event, the fuel would be dumped.

Fuel, Sodium or NaK Leaks

Since leaks between the various fluid systems can give rise to some of the most serious accidents that can be postulated for this reactor system, it is important to examine the conditions which could cause a leak, the reactions which would subsequently occur as a consequence of the leak, and the ultimate hazard. If the many welds are good, as will be determined by radiographic techniques, as well as by preliminary testing, any leaks that might occur would probably result either from corrosion or a fatigue crack. Corrosion is far more probable and is of particular concern, largely because of the uncertainties associated therewith.

The corrosion process is discussed in Appendix C. From the test data included there, it is estimated that the corrosion penetration in the hot zone of the ART would be of the order of 15 to 18 mils if no reductions, in comparison with present experience, can be effected in the corrosion rate. The wall thicknesses to which the fuel is to be exposed will be 125 mils in the core and 25 mils in the heat exchanger. In the core the wall thickness is believed to be ample; in the heat exchanger the metal would sustain a steep temperature gradient because of cooling by the NaK so that the penetrations given above may not apply. However, should component tests fail to justify this assumption, heavier tubing will be used.

If a leak, either from corrosion or fatigue, did occur, various chemical reactions could occur between the fuel and the Na or NaK, depending upon the fuel composition, the size of the leak, and whether the leak was into or out of the fuel system. The consequences of the chemical reactions from each of 16 distinct leak conditions are also discussed in Appendix C. It is apparent that the resulting hazards are dependent upon the assumptions made regarding the leak size, extent of completion of the reaction, and the precipitation of insoluble particles. The most severe case imaginable is discussed in a following section entitled "Accidents Causing Rupture of the Pressure Shell." The probable consequences are, however, much less severe and are discussed below. The action taken in each case would be to dump the fuel and the subsequent hazards would be small.

Fuel Leak in Core. Since the fuel pressure is maintained below that of the sodium, a core leak would most probably result in the addition of sodium to the fuel. The sodium would dilute the fuel mixture and, in the reaction of sodium with the fuel, UF_3 would be produced. The reaction could continue until metallic uranium was produced, which, in turn, would be deposited in the hotter part of the system (i.e., between the core and the heat exchanger). This situation would be handled by dumping the fuel--no hazard would ensue.

On the other hand, if the fuel were to leak into the sodium in the moderator region, the result would be more serious because such a leak would certainly allow excess uranium to be present in the core region. In this event, the fuel would be dumped as soon as possible.

Fuel Leak Into the Heat Exchanger. A leak in the heat exchanger would either admit fuel into the NaK system or vice versa. A fuel leak into the NaK system would increase the activity outside the primary shield, while a NaK leak into the fuel system might result in excess uranium in the core. Either of these situations would be undesirable, but it is now felt that a fuel leak into the NaK system would be the more hazardous; therefore the pressure of the NaK in the heat exchanger will be maintained higher than that of the fuel in the fuel system.

Accordingly, a leak in the heat exchanger would probably result in NaK entering the fuel. As with sodium, UF_3 would be formed and the fuel mixture would be diluted. Eventually uranium would be formed and would be deposited in the hotter section of the system. This situation would be handled by dumping the fuel and no hazard would result.

If the fuel pressure were to be greater than that of the NaK and a leak occurred, some of the fission products would move outside the reactor cell. The radiation doses which would be experienced at various locations in the facility with as much as 1% of the fuel in the NaK system are given above in Table 5.1. The dose rate in the control room 15 min after the shutdown is shown to be only 0.2 rem/hr. The fuel and then the NaK systems would be dumped.

Other Sodium or NaK Leaks. In addition to leaks involving the fuel, the NaK systems might leak externally, and a leak might develop in the Na-to-NaK heat exchanger. A leak of sodium into the NaK would increase the gamma activity of the NaK, while a NaK leak into the sodium system would decrease reactivity in the core, since potassium is more poisonous to the reactor than sodium is. Neither of the above situations presents a serious hazard and would require only that the systems in question be drained.

In the event of an external NaK leak to air, the resulting fire would release some activity (35 curies, maximum). The system would be dumped and the fire would be extinguished.

Sodium or NaK Fires. With regard to fire as a hazard, the building (Figs. 3.1 and 3.2) carries a Uniform Building Code fire rating of 2 hr. Inflammable materials are not used in any appreciable quantity in the construction of the building or the reactor. The reactor, as well as the associated plumbing, pumps, and heat transfer equipment, has been examined rather closely from this standpoint because of the high (up to 1600°F) operating temperatures involved. However, except for the use of Na and NaK as coolants in the system, even the high temperatures (in the absence of combustible material) present no hazard.

The possibilities of a sodium or a NaK leak have been previously discussed. However, during operation, the reactor cell will be filled with helium (or nitrogen) in which sodium and NaK are not flammable. In fact, experiments have shown that even the potentially dangerous reactions of NaK and water are greatly reduced in the absence of oxygen. A NaK leak external to the cell would, however, result in a fire, and therefore the Alkali Metals Area Safety Guide[1] will be employed. Materials which will safely extinguish a NaK or Na fire are graphite powder and Ansul-Met-L-X (sodium chloride coated to prevent the absorption of moisture).[2] Adequate quantities of these materials will be kept at convenient locations.

Since such conventional extinguishers as water, CO_2, and sand should not be applied to a NaK fire, the conventional sprinkler system has been omitted from the design of the building; however, a fire hydrant on a 6-in. main is provided 20 ft from the building.

1. P. L. Hill, Alkali Metals Area Safety Guide, Y-811 (Aug. 13, 1951).

2. A trade compound developed by Ansul Chemical Company, Marinette, Wis.

Accidents Caused by the Natural Elements

As a consequence of the particular location of the ART and the type and material of construction in the building, there is little probability of any damage occurring from natural elements which could create a hazard.

Floods and Earthquakes. Neither floods nor earthquakes present a serious hazard to the ART. As a consequence of the particular topography selected for the site, a flood, and therefore flood damage, is impossible. Further, the data on the frequency and severity of earthquakes in the Oak Ridge area show that the probability of earthquake damage is extremely small (see section on seismology of area in Appendix A, "Characteristics of Site").

Windstorm. With regard to windstorms, the building is designed to the Uniform Building Code criteria. These criteria provide for design against wind loads of 20 lb/ft^2 (about 100 mph) without exceeding the allowable normal working stress of 20,000 psi in the steel structure. A review of the meteorological data for this area[3] shows that it is highly improbable that winds of this magnitude will even be approached at the sheltered site of the ART.

Nuclear Accidents Causing Rupture of the Pressure Shell

It is always instructive to consider the consequences of what might be considered the worst conceivable nuclear accident. Therefore the estimated reactivities from various changes in the reactor are given in Table 5.2. From this table it may be seen that a $\Delta k/k$ of 0.09 is the highest that may be expected. It is then necessary to consider the maximum rate at which this or any other Δk could be introduced into the reactor.

TABLE 5.2. ESTIMATED REACTIVITIES FROM MAJOR CHANGES IN REACTOR

Total value of control rod	0.005 $\Delta k/k$
Control rod motion	0.00033 $(\Delta k/k)$/sec
Temperature coefficient of reactivity	-5.5×10^{-5} $(\Delta k/k)/°F$
Removal of sodium from passages through reflector	+0.0015 $\Delta k/k$
Removal of sodium from passages through island	+0.0005 $\Delta k/k$
Fuel replacing sodium in core shell and reflector cooling passages	0.09 $\Delta k/k$

3. R. L. Myers and J. Z. Holland, A Meteorological Survey of the Oak Ridge Area, ORO-99 (Nov. 1953).

It has been possible to conceive of two extreme situations which appear to establish an upper limit on the rate at which the reactivity could increase. They are as follows: (1) fuel abruptly begins to precipitate out in the core and the fuel stream enters the core at the normal rate but no uranium leaves in the exit stream, and (2) fuel abruptly enters the moderator cooling passages replacing the sodium. A third situation--one in which the beryllium reflector melts and mixes uniformly with the fuel--was examined, but it developed that this accident would not give nearly as large a rate of increase in k_{eff} because the high heat capacity of the reflector would keep the beryllium from melting rapidly. The rate of temperature rise in the portion adjacent to the fuel region would be only 20°F/sec at 60 Mw. The two most severe accidents are discussed below. In addition, the vulnerability of the reactor cell to penetration by pressure shell fragments, if the pressure shell were to rupture as a consequence of other accidents, is examined.

The investigation of these two extreme nuclear accidents may be summarized as follows:

1. The inherent stability of circulating-fluoride-fuel reactors derived from their high negative temperature coefficients make the proposed 60-Mw reactor self-regulating even for extremely rapid changes in k_{eff}.

2. Even if the pressure shell were to be ruptured as a result of an extremely high rate of increase of k_{eff} [40% $(\Delta k/k)$/sec], the pressure shell fragments ejected would not pierce the 1/2-in. wall of the reactor cell.

3. The radiation hazard from all the fuel in the bottom of the reactor cell would be even less than for the case with 1% of the fuel in the NaK system, i.e., only 0.04 rem/hr in the control room 15 min after the accident (see Table 5.1).

Fuel Precipitation in the Core. In the event of a sodium or NaK leak into the fuel system, uranium might be formed which could deposit out in the core (Appendix C). It has therefore been assumed for consideration of this hazard that (1) all the uranium fluoride in the fuel is reduced to free uranium, (2) this uranium precipitates out in the core as fast as it is formed so that no uranium leaves with the exit fuel stream, and (3) the fuel flow (and hence uranium flow) into the core is maintained at the maximum rate.

The consequences of the power surge in this case have been conceived as taking place in the following manner. The power and hence the temperature would increase and cause expansion of the fuel and an increase in the pressure level throughout the fuel circuit. This pressure rise would be propagated with the velocity of sound, i.e., at a high rate compared with the rate of the pressure increase. The fluid pressure would rise until the pressure shell ruptured. Thermal expansion of the fuel would continue until the reactivity was reduced to less than unity and the power dropped back to a low level. Boiling of the fuel might or might not take place.

If boiling does take place, the heat of vaporization of the ZrF_4 is 40 kcal/mole.

This accident has been considered analytically in Appendix D and by a numerical approach in Appendix E. The results of the two methods are reasonably consistent and are summarized in Fig. E.1 of Appendix E. For this type of accident the pressure shell is ruptured only if the reactivity is increased at a rate higher than that corresponding to half the rate at which fuel would be pumped into the core. If the pressure shell should rupture, the reactor fuel and sodium would be spewed into the shield where they would mix and react with each other and with the shield water. The resulting chemical reactions are discussed in Appendix B. As may be seen in this Appendix all the chemical reactions may be contained, with the possible exception of the case discussed in a following section in which there is an air atmosphere in the reactor cell and a hydrogen explosion becomes a possibility.

Fuel in Moderator. If the sodium pressure is not maintained higher than that of the fuel in the core, fuel would enter the moderator region in the event of a core shell leak. It has been assumed that as a consequence of unforseeable events an abrupt rupture of the reflector shell close to the core inlet would begin to discharge fuel into all the cooling passages through the beryllium at a rate equivalent to that given by the sodium velocity. While the cooling passages would probably plug close to their inlets it is interesting to construct the probable course of the accident if it is assumed that no plugging occurs. The initial fuel velocity through the passages would be about the same as that for the sodium, i.e., 30 fps. Because of the higher density of the fuel, its velocity would fall off as it penetrated the reflector. The worst case would be that in which the fuel entered all the coolant passages simultaneously, thus giving the maximum rate of increase in reactivity. The initial fuel velocity through the reflector would give a transit time of about 1/7 sec, or about 1/7 of the corresponding value for the core. The volume of these passages is about 0.16 ft^3, as compared with 3 ft^3 in the core. The increase in k_{eff} if fuel filled the passages in the reflector and island has been computed to be 0.09. The average power density in the fuel in the reflector was calculated to be six times that in the core. This would make the average rate of temperature rise in the reflector fuel about equal to that for the fuel in the core. However, the rate of temperature rise at the nose of the fuel columns in regions of high importance would be at least twice the average so that the rate of temperature rise there should be at least twice the average value in the core.

Calculations were made for this case, except that the rate of increase in k_{eff} was taken to be twice as great to give a truly extreme case. In this instance the power was found to rise to about 6000 Mw in about 80 msec, at which point the fuel in the beryllium would begin to boil. The pressure in the reflector cooling passages would rise abruptly to about 500 psi, the fuel would be expelled from them in about 10 msec, and, at that point, the core fuel would have reached a temperature of about 300°F above normal; the reactor would then be on a 20-msec negative period.

The tabulated calculations for this accident are given in Appendix E, and the results are shown in Fig. E.2. Again, the resulting chemical reactions following a rupture of pressure shell could not give sufficient heat to rupture the reactor cell unless, possibly, oxygen was present in the cell.

Penetrability of Reactor Cell by Pressure Shell Fragments. The pressure shell will be constructed to burst open at a pressure of 1000 psi. The velocity, V, of fluid escaping through a crack if rupture should occur at this pressure can be computed as follows:

$$V = \sqrt{2gH} \quad ,$$

where g is the acceleration and H is the height of a fuel column giving 1000-psi pressure. With a fluid density of 200 lb/ft^3, this height would be

$$\frac{1000\ (lb/in.^2) \times 144\ (in.^2/ft^2)}{200\ (lb/ft^3)} = 720\ ft \ ,$$

and

$$V = \sqrt{2 \times 32.2\ (ft/sec^2) \times 720\ (ft)} = 215\ fps \ .$$

This velocity of 215 fps is certainly a reasonable one and should not give any particular trouble. It is also important because it represents the maximum possible velocity of a fragment that might be broken out of the pressure shell in the event of a hydrostatic rupture. This velocity has to be compared with the velocity required to penetrate the 1/2-in. wall of the enclosure. As discussed in Appendix F, the penetrating power of projectiles varies with their shape, hardness, strength, mass, and velocity, with a 6-in. cast-iron sphere giving a rough standard for this case. As shown in Appendix F, the velocity for penetration of 1/2-in. steel plate by such a sphere is roughly 380 fps. Since the pressure shell is designed to rupture at the bottom, and since the shielded drain tanks for the fuel and sodium will be carried on a substantial floor immediately under the reactor, the tank bottom is well protected from any pressure shell fragments, particularly since only small fragments could have a velocity as high as 215 fps. Thus it is clear that, even in the most pessimistic case, penetration of the cell wall is out of the question, even if no account is taken of the energy loss of the fragment during its travel through the lead and water.

Accidents That Might Rupture the Reactor Cell

It was demonstrated in the preceding section that the reactor cell will not be ruptured as a consequence of even the worst conceivable nuclear accidents. If it is postulated that one such nuclear accident may coincide with the production of all the heat that could possibly be released from the chemical combinations of various materials in the reactor and in an oxygen atmosphere, enough additional energy would be available to give still higher cell pressures and possibly rupture the reactor cell. As pointed out in the previous section, since the chemical reactions are more serious with an oxygen atmosphere, the cell atmosphere will be maintained greater than 99%

nitrogen. The oxygen concentration will be monitored, and the instrumentation will be interlocked (during power operation) to dump the fuel whenever the oxygen concentration is greater than 1%. It then appears that the only means by which both the pressure shell and the reactor cell could be ruptured simultaneously would be as a result of bomb damage. The possibilities of such a bombing, either by sabotage or aerial bombing, are remote. The principle hazard associated with the simultaneous rupture of the reactor pressure shell and cell would be from the dispersion of the activity. This is discussed in the following chapter on "Dispersion of Airborne Activity."

Hydrogen Explosion. In considering the various chemical reactions (see Appendix B), one way that the reactor cell might conceivably be ruptured would be by a hydrogen explosion following the reaction of the sodium or NaK with water in an air atmosphere. As mentioned above, nitrogen is to be maintained within the reactor cell once any appreciable radioactivity has been generated in the fuel. Oxygen absorbers will be exposed to the atmosphere within the cell and an oxygen monitor will be interlocked to open both the fuel and the NaK drain valves if the oxygen concentration goes above 1%.

The energy produced from the various reactions, including those requiring an oxygen atmosphere, have been calculated in Appendix B and are summarized in Table 5.3. The most serious event would be that in which the shield water would combine in stoichiometric proportions with all the sodium and NaK in the system, and the resulting hydrogen would burn in the presence of the available oxygen in an air atmosphere in the reactor cell. If no heat were lost to the cell walls, the pressure in the cell would reach 181 psia. Obviously, this is an overestimate because considerable heat would be removed by the surrounding water during the course of the reaction.

TABLE 5.3. SOURCES OF ENERGY

Heat from reaction of 1000 lb of Na and NaK with water	2.07×10^6 Btu
Heat from reaction of hydrogen with available oxygen in air-filled reactor cell	1.56×10^6 Btu
Heat from reaction of 1000 lb of Na and NaK with air	2.90×10^6 Btu
Heat from reaction of 1200 lb of zirconium-base fuel with sodium	0.98×10^6 Btu
Heat from extreme nuclear accident	0.3×10^6 Btu
Fission-product decay heat emitted during first 2 hr after shutdown (assuming no fission-product removal)	8×10^6 Btu

If all of the hydrogen were to mix with the oxygen without igniting, detonation could take place and give a shock wave that would increase the stresses in the walls of the reactor cell. Of the 181 psia only part, or 66 psia, arises from the hydrogen-oxygen reaction. If it is assumed that the pressure increase associated with detonation is twice the normal pressure rise, or 132 psia, and that the pressure from all reactions prior to the explosion is not relieved by heat removal by the walls of the cell, then a peak pressure of 247 psia would result. This value is about double that for the case in which a nitrogen atmosphere is maintained within the tank.

In all probability the hydrogen would burn as it was formed from the NaK-water reaction. The simultaneous occurrence of several unlikely events would be necessary for an explosion; namely, the oxygen absorbing and monitoring system would fail, an air leak into the cell would occur, a leak between the sodium or NaK and the shielding water would occur, and the hydrogen would not be ignited as it was formed but would react only after the major part of the NaK-water reaction had gone to completion. Even if all of these things did happen, it seems unlikely that the resulting shock wave could rupture the tank wall, particularly in view of the inertia of the steel wall and the water surrounding it.

Damage From High Explosives. Blowing up of the ART with explosive charges set from within by saboteurs would be feasible, as for any installation. However, the effectiveness of such action would depend primarily on the proximity of the charge with respect to the most critical components, i.e., the reactor pressure shell and the fuel drain system. Accessibility to these components and therefore vulnerability will determine the effectiveness of this type of sabotage. The double-walled reactor cell serves as a barrier against entry and would be formidable as protection against external explosions. However, the successful placement of an explosive charge within the container during servicing operations could be effective. In that event, the reactions of the reactor fluids, including the shield water, would take place as described in the preceding section.

If a charge of thousands of pounds of explosives were detonated just outside the reactor cell it would be possible to rupture both the pressure shell and the reactor cell so that the fission product activity in the reactor would be released and a most serious situation would result. Both vessels might also be ruptured by aerial bombing. Either of these cases seems exceedingly improbable. The questions of the strategic importance of the reactor test, its vulnerability because of the double-walled container, and its isolated location in regard to possible hazards to other installations and thickly settled areas must be taken into account. A bombing attack would most certainly be expected under wartime conditions, and appropriate measures could be taken at that time should the eventuality occur.

Effectiveness of the Reactor Cell in Containing Hazards

No allowance for the temperature rise of the atmosphere within the double-walled reactor cell associated with the release of fission-product afterheat has been made. Calculations were made to include this factor as well, and it was found that the temperatures and pressures would fall off slowly after the accident. Thus these values represent maximums; any severe accident would almost certainly involve marked quenching by the shield water, probably to the extent that little pressure and temperature rise in the tank would be experienced.

One of the main reasons for surrounding the inner tank with a thick layer of water was to provide a simple, positive cooling system. The worst heat load likely to be thrown on this cooling system would be that resulting from discharge of fuel into the bottom of the tank after a long period at high power. The lower portion of the tank will be designed so that the fuel will be spread out in a layer 2 in. at the thickest point. For this thickness of fuel and a 1/2-in.-thick steel plate tank bottom, it can be shown that:

Power density in salt from fission-product activity 1 min after 1000 hr at 60 Mw = 0.33 x 300	10 w/cm^3
Heat release rate from 2-in.-thick layer	47,000 w/ft^2 160,000 $Btu/hr \cdot ft^2$
Δt in steel plate	400°F
Δt in water film	40°F
Δt in fuel layer (assuming no convection)	400°F
Area of layer against shell	36 ft^2
Amount of water required to absorb 8×10^6 Btu (heat from fission products in first 2 hr)	
If its temperature rise is 130°F	62,000 lb (1000 ft^3)
If its temperature rise is 130°F and it vaporizes	8,000 lb (32 ft^3)
Amount of water required to absorb a total of 20×10^6 Btu (heat from fission products in next 22 hr)	
If its temperature rise is 130°F	154,000 lb (2500 ft^3)
If its temperature rise is 130°F and it vaporizes	20,000 lb (32 ft^3)

Amount of water in 27-ft-dia outer tank	1,000,000 gal (14,000 ft^3)
Amount of water in 27-ft-dia disk 1 ft thick	4,200 gal (570 ft^3)

Actually, any accident severe enough to dump all the fuel charge into the bottom of the tank would probably also rupture the shield so that the shield water would float on the surface of the fluoride. It would boil violently, and water vapor would rise, condense on the tank walls, and drain back to the bottom of the tank.

The double-walled cell was devised primarily in an effort to give a thoroughly reliable means for absorbing the heat evolved in any accident, no matter how severe. No pumps or other motor-driven equipment would be required and, even if the electrical power supply were to remain inoperable for days after the accident, there would be enough heat capacity in the water so that little, if any, of it would vaporize. The high surface heat transfer coefficient associated with boiling of the water should ensure good cooling of the cell walls.

Comparison of Various Reactor Assembly Containers

Because of the importance of containing the products of a reactor accident, several different potential reactor containers were examined. A comparison of key data for the most promising types of container is given in Table 5.4, and the worst set of conditions applicable to each case is presumed. As may be seen, the reactor assembly cell proposed in this report compares favorably with the hemispherical and ellipsoidal buildings. The double-walled cell also appears superior in that it would be less subject to sabotage. Even if both the inner and the outer tanks were ruptured by sabotage and the reactor melted down, the residue would tend to sink to the bottom of the tank pit where it would be flooded by the water that had filled the region at the top and between the tanks. This water would serve both to absorb the heat of any reaction and as a shield to reduce the radiation level at the top of the pit. After careful review of these and a host of lesser considerations, the 24-ft-dia double-walled cell was chosen as the most promising test facility.

TABLE 5.4. COMPARISON OF HAZARD DATA FOR SEVERAL TYPES OF REACTOR CONTAINERS

	24-ft-Dia Double-Walled Cell with 11-ft Straight Section	200-ft-Dia Ellipsoidal Building	115-ft-Dia Hemispherical Building
Heat released, Btu	2.4×10^6*	10^7	10^7
Container volume, ft^3	12,230	1.2×10^6	0.4×10^6
Container surface area, ft^2	2,640	4.3×10^4	2.1×10^4
Peak gas temperature, °F	2,792	130	220
Peak gas pressure, psig	118	1.4	3.9
Required shell thickness, in. (for allowable stress = 18,000 psi)	1.0	0.15	0.09
Weight of steel in inner shell, tons	36	134**	40

* The fission product afterheat was not included in calculating the peak gas temperature and pressure for the double-walled cell because adequate cooling had been provided.

**Includes steel framing.

6. DISPERSION OF AIRBORNE ACTIVITY[1]

Meteorological data have been used to calculate the possible radiation hazards to the Laboratory and civilian population as a result of both normal and accidental release of radioactive materials from the ART. The normal method of discharging activity will consist of directing the off-gases through a charcoal-filled pipe which will remove most of the activity and effect a more than two-day holdup before ejecting the residual gases up the stack. The stack in carrying the large volumes of heated air produced by the process systems will give very large plume rises in winds under 10 mph. The resultant ground exposures will be always and everywhere substantially below tolerance. In fact, even when there is no process air flow up the stack, the resulting exposures will be below tolerance. The process air will, however, effect a reduction in the resulting ground concentration of around 10, which will be desirable in some emergency situations.

In the calculations for an accidental release, a value of about 6×10^8 cal was used as the minimum amount of heat which could cause all the fission activity to be given off in a gaseous cloud. The total activity present was then assumed to be that given by Mill's formula,[2] even though ARE operation indicated that some of the activity was continuously removed and therefore not available to the disaster. This activity will be safely dispersed from a hot daytime cloud, but it will exceed tolerance at night by a maximum factor of 5 or 17, depending upon which tolerance value is used. As would be expected, the doses from the cold cloud or from rainout of either cloud would be well above tolerance.

It is worthy of note that the assumptions made for the calculations which follow, both for the case of the discharge of activity up the stack and for the dispersion of activity from a disaster, have been conservatively taken at every step. The resulting safety factors combine to effect indicated doses which are at least an order of magnitude higher in most instances and many orders of magnitude higher in other instances than the doses which could reasonably be expected to occur. Since it would become tediously repetitious if this conservatism were to be noted for every situation in the following section, the more pertinent factors are listed and discussed below.

Tolerances. A breathing rate of 30 liters per minute is used, even though this is the breathing rate for an excited man and such a rate cannot be maintained by an individual over a prolonged period. The krypton tolerance of 6.3×10^{-3} curie/m^3 for continuous exposure used in the calculations gives only 300 mrem over the duration of the test (1000 hr). No decay was taken in the krypton activity after it left the holdup system.

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1. Most of this section was written by R. F. Myers and D. R. Purdy of the U. S. Weather Bureau, Oak Ridge, Tennessee.

2. M. M. Mills, A Study of Reactor Hazards, NAA-SR-31, p. 72 (Dec. 7, 1949)

Holdup System. In the charcoal-filled pipe there will be over 100 times the minimum amount of charcoal required to effect a two-day holdup of krypton. In the calculations, no credit is taken for residence time in the reactor system or travel time in the off-gas system.

Meteorological Parameters. A stability factor (nighttime) of 0.4 was used in the disaster calculation rather than the more likely value of 0.35. The hot cloud and the stack plume rises were limited in some instances, even though greater heights would have effected greater dispersions. Although the nighttime stable conditions will not last much longer than 16 hr, the unrealistic assumption of continuous nighttime conditions was not used for continuous exposures or for disaster-cloud travel times of greater than about 16 hr. Low average wind velocity values were used for the hot cloud, even though equally justifiable higher values would have effected greater dispersions. Also, for each calculation that gives the maximum dose at any location, it is assumed that the wind always blows in that direction with a constant optimum value.

Fission Products in Hot Cloud. The amount of heat in the hot cloud is a lower limit for the amount of heat required to vaporize all the fuel because the heat of vaporization of the first component which comes off (as the fuel is heated) is used for determining the heat required to vaporize all the fuel. For determining the activity in the cloud, an upper limit is used, since all the fuel would never be in the right place at the right time to be vaporized. Also, operation of the ARE indicated that some activity may be removed continuously.

Radiation Tolerances

The maximum total dose which the civilian population should be permitted to receive in any accident is 25 rem. The maximum permissible exposure to contaminated atmosphere which will give a dose of 25 rem by inhalation is therefore of considerable interest. A value for the maximum permissible exposure of 10 curie·sec/m^3 has been given by Marley[3] for total fission products. However, Marley used a breathing rate of the order of 6 liters/min that is considerably lower than that for the average excited man, which is around 30 liters/min. In applying Marley's tolerance to a given condition the total radiation is assumed to decay according to $t^{-0.2}$ as the radioactive cloud moves out from the source.

On the other hand, T. J. Burnett of the ORNL Health Physics Division has calculated,[4] on the basis of a selected group of 30 long-lived fission products, that the maximum permissible exposure to these isotopes is 1.44 curie·sec/m^3 after 39 days[5] of reactor operation. The 30 isotopes

3. W. G. Marley, Health Physics Considerations in a Reactor Accident, R/SAF/WK/3 (no date).

4. Appendix G, "Exposure Hazard Calculations," this report.

5. Thirty-nine days is used here (rather than 41, which would be a closer approximation to the anticipated 1000 hr of operation time) because some calculations were already available with this time and the difference is small.

selected define the limiting tolerance for all fission products. After 1000 hr of operation, these 30 isotopes represent about 8% of the total activity present at 1 sec, if $t^{-0.2}$ is assumed for all the activity. Since these are long-lived isotopes, no decay correction is applied. (It is further shown in Appendix G that this group of 30 isotopes may be reduced to a group of six isotopes which contribute over 95% of the dose to the bone.)

As was noted previously the off-gas system is designed to routinely discharge less than 0.001 curie/sec of Kr^{88} to the stack. The tolerance for Kr^{88}-Rb^{88} has been calculated[6] to be 6.3×10^{-8} curies/m^3 for continuous exposure. This calculation permitted a dose rate of 0.3 mrem/hr for the 1000 hr of contemplated reactor operation.

Discharge of Activity up the Stack

The ART stack is 78 ft high and 10 ft in diameter. Its dimensions were largely defined by the air flow from the process systems since it was desired to employ this air in order to effect greater dispersion of the off-gas. The design capacity of the air blowers is 3×10^5 cfm of air at ambient temperature. This volume of air when heated in the hot-air radiators to 750°F expands to 6.7×10^5 cfm, and therefore, during power operation, this latter quantity of air is discharged up the stack.

The dispersion of activity from the stack has been considered for three conditions of air flow: (1) 6.7×10^5 cfm of 750°F air, (2) 3×10^5 cfm of ambient air, and (3) no air flow. With the two-day holdup provided in the off-gas system before the off gases reach the stacks, the off gases may be discharged without anywhere exceeding the Kr^{88} tolerance of 6.3×10^{-8} curie/m^3 for continuous exposure for the first two conditions. For no stack air flow, the Kr^{88} tolerance will be exceeded by a factor of 4. Furthermore, if it becomes necessary following a contained disaster to dispose of large amounts of activity, as much as 8 curies/sec could be discharged up the stack with 3×10^5 cfm of ambient air and still not exceed an internal dose rate of 1 rem/hr.

Normal Operation. During normal operation the Aircraft Reactor Test will produce about 6.7×10^5 cfm of air at 750°F from the air radiators. This air will be ejected up the stack and hence will aid in the dispersal of the gaseous fission products which will be given off by the hot fuel. The work of Davidson[7] (which has been compared with actual smoke observations[8] from a large TVA steam plant stack for verification) has been used

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6. Private communication from T. J. Burnett.

7. W. F. Davidson, "The Dispersion and Spreading of Gases and Dusts from Chimneys," Ind. Hyg. Foundation Amer., Trans. Bull. No. 13 (1949).

8. F. W. Thomas, TVA, Wilson Dam, Ala., Plume Observations, Watts Bar Steam Plant (1952), unpublished manuscript.

in estimating the plume rise expected from a 10-ft-dia stack, 78 ft high. Figure 6.1 shows estimated plume rises versus wind speed for both the 6.7×10^5 cfm air at 750°F and 3×10^5 cfm of air at ambient temperature.

The calculated plume rise values were used, together with appropriate meteorological data derived from the observations reported by Myers and Holland,[9] to compute the maximum ground concentrations and the distance from the stack of the maximum ground concentration under a wide range of wind speeds. The full plume rise was used in the unstable or daytime case, but the plume rise was limited to about 600 meters at night, which corresponds to the upper limit of observed rises from the large TVA steam plant stacks during stable conditions. The calculated rises below the 600-meter level were used for the nighttime rise. This treatment of the stable case minimizes the safety factor of the light-wind stable case which might be the most doubtful. The important dispersion conditions which are considered are those which occur with winds under 10 mps and are representative of 99.9% of the hours of wind observed at the site. The parameters used for these calculations are given in Table 6.1.

The results of these calculations for steady emission of 0.001 curie/sec are given in Tables 6.2 and 6.3, with and without the decay correction. The decay correction given here is that determined by the decay rate for equilibrium reactor fission products, $t^{-0.2}$.

As previously noted (section on "Radiation Tolerances") the concentration of Kr^{88} for continuous exposure is 6.3×10^{-8} curie/m^3. Furthermore, for a reactor operating at 60 Mw the equilibrium discharge rate of Kr^{88} after a two-day holdup in the off-gas system is 0.0009 curie/sec (section on "Off-Gas System"). The maximum concentration (with no decay correction) occurs during the day at 0.37 mile from the source when the wind speed is 10 meters per second (1 mps = 2.24 miles per hour). This concentration (6.08×10^{-9} curie/m^3 per 0.001 curie/sec emitted) is a factor of 10 below the tolerance for continuous exposure. Furthermore, it should be noted that the wind will not blow in the same direction and with the same high (10 mps) velocity for 1000 continuous hours of daytime conditions, and, in addition, the actual off-gas holdup system will have over 100 times the amount of charcoal required to effect the two-day holdup which was used in the above calculations.

Operation Without Heating Stack Air. Except when the reactor is operating at full power the air flow up the stack will be less than 6.7×10^5 cfm and the air temperature will be less than 750°F. For the limiting case with no power being removed from the system, there will be 3×10^5 cfm of ambient temperature air flow up the stack. A by-pass air duct will be provided around the radiators so that the air may be sent up the stack without cooling the reactor. The lower temperature and smaller air flow will produce the lower stack rises which are given by the lower curve in Fig. 6.1. These plume rises, together with the meteorological data derived from ORO-99, were

9. R. L. Myers and J. Z. Holland, A Meteorological Survey of the Oak Ridge Area, ORO-99 (Nov. 1953).

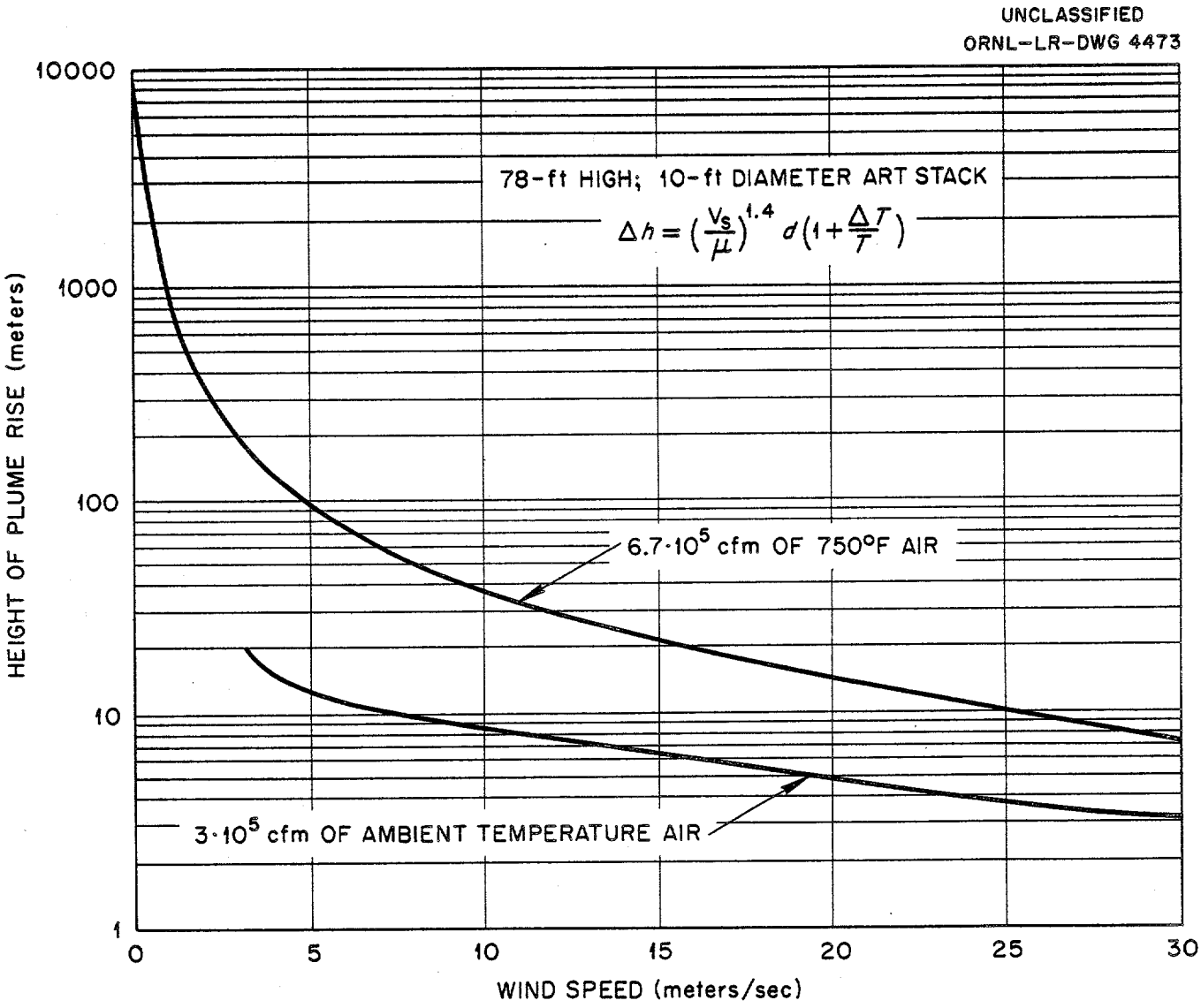

Fig. 6.1 Height of Plume Rise from ART Stack.

TABLE 6.1. METEOROLOGICAL PARAMETERS FOR DETERMINING GROUND CONCENTRATION OF GASES RELEASED FROM STACK

Parameter*	Value of Parameter Relative to Windspeed							
	During Nighttime				During Daytime			
For 6.7×10^5 cfm of Air at 750°F								
Wind speed (mps)	1	2	4	10	1	2	4	10
c^2	3.6×10^{-3}	3.6×10^{-3}	5.6×10^{-3}	9.0×10^{-3}	1×10^{-2}	2×10^{-2}	2.9×10^{-2}	4.4×10^{-2}
x_o (meters)	515	325	163	63	182	83	46	21
$h + \Delta h$ (meters)	550	379	159	62	961	379	159	62
n	0.35	0.35	0.35	0.35	0.23	0.23	0.23	0.23
For 3.0×10^5 cfm of Air at Ambient Temperature								
Wind speed (mps)	1	2	4	10	1	2	4	10
c^2	5.0×10^{-3}	6.4×10^{-3}	8.1×10^{-3}	1.7×10^{-2}	3.1×10^{-2}	3.1×10^{-2}	4.2×10^{-2}	4.8×10^{-2}
x_o	340	215	186	164	86	61	51	48
$h + \Delta h$ (meters)	220	99	53	33	220	99	53	33
n	0.35	0.35	0.35	0.35	0.23	0.23	0.23	0.23

* c is the diffusion parameter; x_o is the distance correction; Δh is the correction for h is the height of rise of the plume; and n is the stability factor.

TABLE 6.2. GROUND CONCENTRATION OF GASES RELEASED FROM STACK
(Without Decay Corrections)

Air Flow: 6.7×10^5 cfm at 750°F

Distance from Stack (miles)	Concentration* (curies/m^3) at Wind Speeds of 1 mps	2 mps	4 mps	10 mps
		During Daytime		
0.31			2.77×10^{-13}	5.69×10^{-9}
0.62		7.67×10^{-17}	4.84×10^{-10}	4.65×10^{-9}
1.86	2.24×10^{-33}	8.11×10^{-11}	2.10×10^{-9}	1.05×10^{-9}
4.35	7.54×10^{-15}	7.76×10^{-10}	7.45×10^{-10}	2.20×10^{-10}
6.24	2.80×10^{-12}	7.27×10^{-10}	4.17×10^{-10}	1.20×10^{-10}
12.48	1.62×10^{-10}	3.92×10^{-10}	1.62×10^{-10}	1.53×10^{-11}
31.1	1.98×10^{-10}	7.84×10^{-11}	2.74×10^{-11}	7.10×10^{-12}
37.28	1.63×10^{-10}	5.62×10^{-11}	1.95×10^{-11}	5.00×10^{-12}
		During Nighttime		
0.31				9.18×10^{-13}
0.62			2.30×10^{-24}	8.53×10^{-10}
1.86		1.72×10^{-33}	2.40×10^{-11}	5.90×10^{-9}
4.35	5.25×10^{-23}	1.87×10^{-15}	1.75×10^{-9}	2.62×10^{-9}
6.24	1.82×10^{-16}	1.4×10^{-12}	2.31×10^{-9}	1.59×10^{-9}
12.48	2.28×10^{-11}	3.26×10^{-10}	1.53×10^{-9}	2.50×10^{-10}
31.1	7.21×10^{-10}	7.70×10^{-10}	4.65×10^{-10}	1.25×10^{-10}
37.28	7.76×10^{-10}	6.8×10^{-10}	3.39×10^{-10}	9.12×10^{-11}

* For unit emission of 0.001 curie/sec.

TABLE 6.3. GROUND CONCENTRATION OF GASES RELEASED FROM STACK

(With Decay Correction, $t^{-0.2}$)

Air Flow: 6.7×10^5 cfm at 750°F

Distance from Stack (miles)	Concentration* (curies/m^3) at Wind Speeds of			
	1 mps	2 mps	4 mps	10 mps
		During Daytime		
0.31			1.07×10^{-13}	2.60×10^{-9}
0.62		2.24×10^{-17}	1.62×10^{-10}	1.86×10^{-9}
1.86	4.50×10^{-34}	1.87×10^{-11}	5.35×10^{-10}	3.40×10^{-10}
4.35	1.29×10^{-15}	1.52×10^{-10}	1.67×10^{-10}	5.95×10^{-11}
6.24	4.43×10^{-13}	1.32×10^{-10}	8.71×10^{-11}	3.02×10^{-11}
12.48	2.24×10^{-11}	2.31×10^{-11}	2.96×10^{-11}	7.80×10^{-12}
31.1	2.28×10^{-11}	1.03×10^{-11}	4.13×10^{-12}	1.28×10^{-12}
37.28	1.82×10^{-11}	7.4×10^{-12}	2.83×10^{-12}	8.90×10^{-13}
		During Nighttime		
0.31				4.20×10^{-13}
0.62			7.73×10^{-24}	3.41×10^{-10}
1.86		3.98×10^{-34}	6.39×10^{-12}	1.90×10^{-9}
4.35	9.00×10^{-24}	3.67×10^{-16}	3.93×10^{-10}	7.08×10^{-10}
6.24	2.88×10^{-14}	2.66×10^{-13}	4.83×10^{-10}	4.00×10^{-10}
12.48	3.15×10^{-12}	1.08×10^{-10}	2.80×10^{-10}	1.19×10^{-10}
31.1	8.31×10^{-11}	1.02×10^{-10}	7.04×10^{-11}	2.29×10^{-11}
37.28	8.70×10^{-11}	8.64×10^{-11}	4.92×10^{-11}	1.61×10^{-11}

* For unit emission of 0.001 curie/sec.

then used to calculate the resulting ground concentration at various distances for several wind speeds during both daytime and nighttime conditions. The meteorological parameters used for these calculations are given in Table 6.1. The results of these calculations for a steady emission of 0.001 curie/sec (without the decay correction) are given in Table 6.4.

TABLE 6.4. GROUND CONCENTRATION OF GASES RELEASED FROM STACK

(Without Decay Correction)

Air Flow: 3.0×10^5 cfm at Ambient Temperature

Distance from Stack (miles)	Concentration* (curies/m3) at Wind Speeds of 1 mps	2 mps	4 mps	10 mps
		During Daytime		
0.31	1.18×10^{-22}	1.85×10^{-9}	2.09×10^{-8}	1.34×10^{-8}
0.62	9.86×10^{-13}	1.12×10^{-8}	1.27×10^{-8}	5.35×10^{-9}
1.86	3.85×10^{-9}	5.71×10^{-9}	2.48×10^{-9}	8.85×10^{-10}
4.35	3.72×10^{-9}	1.49×10^{-9}	5.80×10^{-10}	2.04×10^{-10}
6.21	2.27×10^{-6}	8.05×10^{-10}	3.13×10^{-10}	1.09×10^{-10}
12.42	8.53×10^{-10}	2.40×10^{-10}	9.24×10^{-11}	3.22×10^{-11}
31.1	1.77×10^{-10}	4.85×10^{-11}	1.74×10^{-11}	6.34×10^{-12}
37.28	1.29×10^{-10}	3.54×10^{-11}	1.29×10^{-11}	4.60×10^{-12}
		During Nighttime		
0.31		5.94×10^{-21}	2.16×10^{-10}	2.53×10^{-10}
0.62		1.23×10^{-12}	7.38×10^{-9}	1.69×10^{-8}
1.86	7.25×10^{-14}	6.62×10^{-9}	1.82×10^{-8}	1.49×10^{-8}
4.35	9.59×10^{-10}	1.09×10^{-8}	7.32×10^{-9}	5.05×10^{-9}
6.21	3.09×10^{-9}	8.70×10^{-9}	4.39×10^{-9}	2.97×10^{-9}
12.42	4.69×10^{-9}	3.47×10^{-9}	1.52×10^{-9}	9.90×10^{-10}
31.1	1.88×10^{-9}	8.42×10^{-10}	3.40×10^{-10}	2.23×10^{-10}
37.28	1.45×10^{-9}	6.34×10^{-10}	2.54×10^{-10}	1.65×10^{-10}

* For unit emission of 0.001 curie/sec.

By extrapolating the data in Table 6.4, it may be shown that the maximum ground concentration of 2.43×10^{-8} curie/m^3 is attained during the daytime with a wind speed of 6.6 mps and at 0.22 mile. Even so, this maximum ground concentration is a factor of 3 below tolerance for continuous exposure--again assuming that the wind always blows in the same direction and at the same velocity.

Operation With No Stack Air Flow. Should it become necessary to discharge the off-gases up the stack with no process air flow, the ground concentrations that will be obtained during the daytime and nighttime conditions are given in Table 6.5 for average wind speeds during the daytime and the nighttime of 2.3 and 1.5 mps, respectively, for which the diffusion parameters are 0.005 and 0.001, respectively.

TABLE 6.5. GROUND CONCENTRATION OF GASES RELEASED FROM STACK
(No air flow and no decay correction)

Distance (miles)	Nighttime Concentration* (curies/m^3)	Daytime Concentration* (curies/m^3)
0.125	2.2×10^{-11}	1.6×10^{-7}
0.37	1.6×10^{-7}	5.8×10^{-8}
0.62	2.5×10^{-7}	2.6×10^{-8}
0.89	2.1×10^{-7}	1.4×10^{-8}
5	2.3×10^{-8}	6.8×10^{-10}
10	7.9×10^{-9}	2.0×10^{-10}
15	4.1×10^{-9}	9.7×10^{-11}
20	2.6×10^{-9}	5.8×10^{-11}
50	6.0×10^{-10}	1.2×10^{-11}
100	2.0×10^{-10}	3.4×10^{-12}

* For unit emission of 0.001 curie/sec.

The highest ground concentration, 2.5×10^{-7} curie/m^3, which occurs at 0.62 mile during the nighttime, is above the continuous tolerance of 6.3×10^{-8} curie/m^3. This condition is amenable to routine reactor operation, since it would never be permitted on a continuous basis nor would the meteorological conditions which combine to produce this peak concentration at that particular location be attained continuously.

Operation With Holdup System By-Passed. An emergency exhaust line will be provided to the interior of the reactor cell for venting the fission-product gases up the stack in the event of a leak in the off-gas holdup system or in the event of some failure which would allow fission-product gases to enter the cell. If such an operation were contemplated on a continuous basis, the criteria for the amount of activity discharged would have to be determined by the method used in the preceding sections. If it becomes necessary, in some emergencies to dispose of large amounts of activity, much more than 1 curie/sec could be sent up the stack without exceeding an internal dose rate of 1 rem/hr. The exact discharge rate may be obtained by spreading out the 1.44 curie·sec/m^3 (Burnett's tolerance for the 30 isotopes) over a 24-hr period, with the result that a continuous exposure of 1.6×10^{-5} curie/m^3 will give dose rates of 1 rem/hr. Since, for example, the maximum concentration with 3×10^5 cfm of ambient air up the stack is 2.43×10^{-5} curie/m^3 per curie/sec emitted, 8% of which is applicable to the above tolerance, then $(1.6 \times 10^{-5})/(0.08 \times 2.43 \times 10^{-5})$ or 8.3 curies/sec will give a maximum dose rate of 1 rem/hr. If only the gaseous activity was used as the source strength and a decay factor for this activity was included, the resulting dose rates would be substantially smaller.

If still larger amounts of activity must be disposed of, reference will be made to the calculated concentration appropriate to the current meteorological condition to determine the feasibility of opening the stack line. However, at any given distance the ground concentration (in curie/m^3) from a continuous source (curie/sec) is numerically identical to the total integrated dose (in curie·sec/m^3) from an instantaneous source (curies). Again by using the tolerance of 1.44 curie·sec/m^3 for 8% of the activity, the data from Tables 6.1 through 6.4 may be used to determine the maximum allowable curies in a puff. Therefore, for example, the highest ground concentration with 3×10^5 cfm of ambient air up the stack is 2.43×10^{-5} curie/m^3 per curie/sec emitted, which is identical to 2.43×10^{-5} curie·sec/m^3 per curie. Therefore, $1.44/(0.08 \times 2.43 \times 10^{-5})$ or 7.4×10^5 curies may be sent up the stack in a single puff. Again, the value was obtained by assuming all fission activity to be present and that no decay had occurred; also the wind speed was assumed to be constant.

Smoke Tracking of the Stack Plume. In order to provide some assurance that the stack is able to handle the high concentrations for which it is designed and to provide a visible trace of the radioactive cloud during operation if the design capacity is exceeded, a smoke generator of the oil-fog type will be included in the off-gas system. Estimates of the required capacity have been made to ensure that the cloud will be visible at adequate distances. The distance at which the output of a 50-gph generator will be visible are shown in Table 6.6.

TABLE 6.6. LIMIT OF VISIBILITY OF A SMOKE PLUME FROM A 50-gph GENERATOR

Wind Speed (mps)	Visibility (miles)	
	Unstable (Daytime)	Stable (Nighttime)
0.5	23.8	116
1	12.0	49
2	4.4	21
4	1.3	7.3
10	0.4	2.2

The minimum visible smoke density per unit cross section area in the line of sight is taken to be 3.3×10^{-2} g/m^2 or 1/10 of the accepted screening level of 0.33 g/m^2. The calculation is based on an integration of Sutton's equation as given in ORO-99.

In the event of an unexpected release or a release larger than the design capacity of the off-gas system, the smoke generator will provide a guide for monitoring and will also act as a visible warning to the Laboratory personnel. The long travel time required to reach much greater distances would not permit the visible smoke to reach any of the large population areas.

Discharge of Activity Following a Disaster

If a disaster occurred in which sufficient energy was evolved to rupture the reactor cell, as well as the pressure shell, the fission activity would be released to the atmosphere, and therefore the hazards in terms of radiation doses resulting from such a disaster have been calculated. In this calculation it was further assumed that there was sufficient heat available to the disaster to vaporize all the fuel (including that external to the reactor core) and thus place the activity in a hot cloud. (in actuality, if the pressure shell does rupture, most of the fuel will leak into the reactor cell where, because of its high melting point, it will rapidly solidify.) With the activity in gaseous form, the dispersion of the resulting cloud can be determined and the total integrated doses at any distance from the site can be calculated. Even if the building is not ruptured the building exhaust fans will effect a complete turnover of the building air in less than 5 min and the resulting radioactive cloud may be regarded as the hot cloud case. With the hot cloud the activity will always be safely dispersed during the day, but at night the activity will exceed the 25 rem tolerance factor of 17 or 5, depending upon which tolerance is used. A calculation has also been made for the cold cloud case.

Minimum Heat Liberated in a Disaster. An attempt was made to calculate the minimum amount of heat that would be required to get all the fission activity into a vapor cloud. These calculations invariably involve a number of assumptions and approximations which, for the purposes of this calculation, have been conservatively taken. The calculation has been made for both the zirconium-base and the alkali-metal-base fuel mixtures. The vapor pressure for ZrF_4 above the $NaF-ZrF_4-UF_4$ (53-43-4 mole %) fuel is represented by

$$\log P = \frac{7160}{T} + 7.37 ,$$

where P is in mm Hg and T is in °K. The boiling temperature is then found to be 2246°F. The heat of vaporization of ZrF_4 from the above mixture is 40 kcal/mole. Since the ZrF_4 is the first constituent to be vaporized, its heat of vaporization represents a lower limit for the fuel. The heat required to vaporize the fission products is the heat required to reach the boiling point plus the heat required to vaporize all the fuel, as determined here by using the heat of vaporization of the ZrF_4. The total fluoride volume of 5.64 ft^3 is used for this determination because the fission activity is distributed throughout the fuel.

The heat required to boil the fuel is given by

$$\phi \text{ (Btu)} = V\rho c \, (\Delta T) ,$$

where ΔT is the temperature rise in °F (950), c is the heat capacity in Btu/lb·°F (0.25), and ρ is the density of the fuel (185 lb/ft^3). The heat required to reach boiling is 2.48×10^5 Btu, or approximately 0.63×10^5 kcal.

The heat required to vaporize the fuel is

$$\phi \text{ (kcal)} = V\rho H_v N ,$$

where H_v is the heat of vaporization (40 kcal/mole) and N is the number of moles per pound. The heat required to vaporize the fuel is 5.24×10^5 kcal and therefore the total energy required to vaporize the fission products is 5.9×10^5 kcal.

In the $NaF-KF-LiF-UF_4$ fuel system the UF_4 is the first constituent to vaporize. The temperature at which UF_4 boils is 1388°C, and the heat of vaporization is 66 kcal/mole.[10] The heat required to raise this fuel to the boiling point is

$$\phi \text{ (Btu)} = 5.64 \times 185 \times 1235 \times 0.4$$

$$= 5.1 \times 10^5 \text{ Btu (or } 1.3 \times 10^5 \text{ kcal)} ,$$

10. A. D. Ryon and L. P. Twichell, Vapor Pressure and Related Physical Constants of UF_4, H-5.385.2 (July 25, 1947).

and the heat required to vaporize it is

$$\phi \text{ (kcal)} = 5.64 \times 185 \times 66 \times \frac{2205}{311}$$

$$= 4.8 \times 10^5 \text{ kcal} .$$

Therefore the total heat required to boil and then vaporize the NaF-LiF-KF-UF_4 fuel is 6.1×10^5 kcal, i.e., slightly greater than that for the zirconium-base fuel. An average value of 6×10^5 kcal is used in the following calculations.

Having determined a value for the minimum amount of heat to be released in a disaster, it is then necessary to estimate the resulting temperature rise of the building air in order to determine the height of rise of the hot gases. The volume of the enlarged building is approximately 48×10^5 ft^3 (1.36×10^4 m^3). Since air has a density of 1.2 g/m^3 and a specific heat of 0.25, the temperature of the air will be increased 265°F (147°C) by the 6×10^5 kcal of heat required for vaporization of the fission products in the fuel.

Height of Cloud Rise. The height of rise of a heated cloud may be calculated from work by O. G. Sutton,[11,12] and, from the height of rise, the volume of the cloud may be derived. Sutton's basic equation gives an approximate solution for the temperature excess of a rising hot puff that originates in an instantaneous point source of heat and mixes with its environment by eddy diffusion. Satisfactory verification of the equation was obtained with published data on the Trinity explosion. The equation is

(1) $$\Delta\theta_c = \frac{Q_H}{2 c_p \rho \pi^{3/2} C^3 z^{3m/2}} ,$$

where $\Delta\theta_c$ = the difference (in °C) between the average potential[13] temperature of the cloud and the potential temperature of the atmosphere, which is assumed to be constant;

Q_H = the total heat liberated (in cal) and transferred to the building air;

c_p = the specific heat of air at constant pressure (0.25 cal/g·°C);

ρ = the air density (1.2×10^3 $g/in.^3$);

C = the virtual diffusion coefficient (allows for natural turbulence of the air and the enhanced turbulence introduced by the hot puff; a value of 0.3 is used here because the calculation is for a relatively stable thermal stratification);

z = the height above the source (in meters);

11. O. G. Sutton, Weather, p.108 (Apr. 1947).

12. O. G. Sutton, The Diffusion of Matter from an Explosion, p. 6, PLP-11 (Dec. 14, 1946).

13. Potential temperature is the actual temperature of the air if it is compressed adiabatically to 1000 millibars.

m = a parameter that results from the increase in size of the effective diffusion eddies during the spreading of the cloud and is estimated by Sutton to be 1.75, in the average case.

In order to determine the level in a stable atmosphere at which the cloud reaches thermal equilibrium, it is assumed that the vertical gradient of potential temperature in the atmosphere is a constant, θ_a', and that at the equilibrium level the temperature excess of the cloud with respect to a neutral atmosphere is just balanced by the actual increase in potential temperature of the atmosphere with height. Then,

(2) $$\Delta\theta_c = \Delta\theta_a = z_{max}\,\theta_a'$$

and

(3) $$z_{max} = \left(\frac{Q_H}{2\,c_p\,\rho\,\pi^{3/2}\,c^3\,\theta_a'}\right)^{2/(3m+2)} .$$

In the stable case, for $\theta_a' = 0.02°C$/meter, which is an average nighttime value for the Oak Ridge area, the calculated cloud rise is about 225 meters. Since, from Eq. 2, $\Delta\theta_c$ is equal to 4.5°C, the volume of the nighttime cloud may be approximated by assuming that the decrease in temperature excess is accounted for by corresponding increase in volume, i.e.,

(4) $$\frac{\Delta\theta_2}{\Delta\theta_1} = \frac{V_1}{V_2} .$$

The nighttime cloud volume at 225 meters would therefore be 4.45×10^5 m^3, as determined by using the value arrived at above of 1.36×10^4 m^3 for the source volume with an excess temperature of 147°C.

In the unstable case (daytime), it is assumed that the cloud levels off at about 1500 meters (300 to 500 meters above the daytime natural cloud base). Then, according to Sutton;s formula (Eq. 1), the temperature excess would be

(5) $$\Delta\theta_3 = \Delta\theta_2 \left(\frac{z_2}{z_3}\right)^{2.62} ,$$

which gives a value of 0.032°C. The daytime cloud volume may then be determined (in the same manner as the nighttime cloud), and is found to be equal to 6.25×10^7 m^3.

Fission Products in the Hot Cloud. The activity contained in the reactor after a normal shutdown, if it is assumed that none of the fission products has been removed, may be calculated from a formula by Mills,[2]

(6) $$A = \frac{11.15P}{t^{0.2}} ,$$

where A = activity (in curies),
P = power (in watts),
t = time (in seconds).

For a reactor operating at 60 Mw, the equilibrium activity 1 sec after the catastrophe would be 6.6×10^8 curies, if an average energy of 1 Mev is assumed. However, Mill's formula contains no augmentation from the additional activity produced by a power excursion during a run-away type of catastrophe. If it is assumed that the additional activity produced during the catastrophe is that required to effect the minimum heat release,

(7) $$A \text{ (curies)} = \frac{Q_H \text{ (cal)} \times 4.18 \left(\frac{\text{watt/sec}}{\text{cal}}\right) \times 3 \times 10^{10} \left(\frac{\text{fissions}}{\text{watt.sec}}\right) \times \frac{1.6}{t^{1.2}} \left(\frac{\text{Mev}}{\text{sec.fission}}\right)}{3.7 \times 10^{10} \left(\frac{\text{Mev}}{\text{sec.curie}}\right)}.$$

Since $Q_H = 6 \times 10^8$ cal, the activity from the excursion is 32×10^8 curies[14] at 1 sec after the catastrophe. Although at 1 sec this activity is of the same order as that in the reactor at equilibrium, it decays according to $t^{-1.2}$ so that after a few seconds it is insignificant as compared with the activity in the reactor, which decays according to $t^{-0.2}$. Consequently, the calculations of ground exposure from the hot cloud need consider only the activity initially in the reactor. Furthermore, it is assumed that only 50% of the equilibrium activity, or 3.3×10^8 curies, is present in the cloud following the disaster.

Internal Exposures from the Hot Cloud

Sutton has also developed a general equation for isotropic diffusion from an instantaneous point source.[15] In the ARE Hazards Summary Report,[16] this equation was developed to provide not only the maximum dose at every point as the cloud moves out but, also, the integrated exposure at the ground for the entire time of the cloud as found by integrating the concentration equation. These calculations have now been further extended by allowing for the decay in the activity of the cloud as it moves out.

In the total reactor tragedy which has been postulated, 2.4×10^6 Btu or 6×10^8 cal represents the minimum amount of heat which would vaporize the 5.64 ft^3 of fuel and hence make the total stored activity in the reactor available in airborne pollution. The height of rise has been calculated (in the preceding section) for a cloud of hot air containing the fission products in the 1.36×10^4 m^3 volume of the building with the addition of the 6×10^8 cal of heat. The basic meteorological data used in these estimates are taken from previous studies reported by Myers and Holland.[9] Table 6.7 summarizes this data.

14. This value, as determined here, is higher than that which could be produced in a excursion as determined from nuclear considerations in Appendixes D and E.

15. O. G. Sutton, Proc. Roy. Soc. (London) 135A, 155 (1932).

16. J. H. Buck and W. B. Cottrell, Aircraft Reactor Experiment Hazards Summary Report, ORNL-1407 (Nov. 24, 1952).

TABLE 6.7. METEOROLOGICAL PARAMETERS FOR TOTAL REACTOR TRAGEDY

Parameter	Definition	Average Value of Parameter	
		Stable, Night	Unstable, Day
h	Height of rise	225 meters	1500 meters
c	Diffusion parameter	0.055	0.114
n	Stability	0.40	0.23
V_o	Initial cloud volume	4.45×10^5 m^3	6.25×10^7 m^3
x_o	Distance correction	4170 meters	5250 meters
u	Wind speed (average)	5 mps	8 mps

The total integrated internal doses with no decay and with $t^{-0.2}$ decay and as calculated by both Burnett and Marley (see preceding sections on radiation tolerance) are given in Table 6.8. As was shown previously, the total activity in the hot cloud is 3.3×10^8 curies. Since the tolerance, as derived by Marley's work, is based on total fission products, it is consistent to permit this activity to decay as the cloud moves out. However, the tolerance of 1.44 curie·sec/m^3, as derived by Burnett (Appendix G), is based on 30 isotopes which represent only 8% of the total and hence the source strength is only 2.65×10^7 curies; no decay may be applied since the isotopes are relatively long lived (> 12 days). Consequently, column 4 of Table 6.8 was obtained by multiplying column 2 by 2.65×10^7 curies and column 5 was obtained by multiplying column 3 by 3.3×10^8 curies. Columns 6 and 7 were then obtained by dividing the numbers in columns 4 and 5 by their respective tolerances, i.e., 1.44 and 10 curie·sec/m^3. Although columns 6 and 7 permit a comparison of the overdoses according to the two tolerances, they are, strictly speaking, not comparable because of the factor of 5 lower breathing rate employed by Marley. It is of interest, however, that the highest dose is only 5.3 (Marley) or 17 (Burnett) times above the maximum. Furthermore, if the Marley value is multiplied by 5, to give comparable breathing rates, the agreement between the two tolerances is exceptionally good.

External Exposure from the Hot Cloud

The dosages that would result from external radiation as the hot cloud passed overhead are given in Table 6.9. The method devised by Waterfield[17] and the units and the conversion factor furnished by S. Visner were used for the calculations. In Waterfield's equation, the total activity was in terms of number of fissions, and it contained a decay factor of $t^{-1.2}$. Visner expressed the activity in terms of reactor power and substituted a decay

17. R. L. Waterfield, Cloud Dosage Calculations, XDC 54-4-12 (Apr. 1954).

TABLE 6.8. TOTAL INTEGRATED INTERNAL DOSES (TID) FROM HOT CLOUD

(1) Distance (miles)	(2) Integrated Dose per Curie, no decay (curie·sec/m^3)	(3) Integrated Dose per Curie, $t^{-0.2}$ decay (curie·sec/m^3)	(4) TID (Burnett) (curie·sec/m^3)	(5) TID (Marley) (curie·sec/m^3)	(6) Overdose (units of 25 rem) (Burnett)	(7) Overdose (units of 25 rem) (Marley)
			Nighttime Average			
0.37	1.5×10^{-12}	3.92×10^{-13}	4.0×10^{-5}	12.9×10^{-5}	2.8×10^{-5}	1.3×10^{-5}
0.89	1.76×10^{-10}	4.29×10^{-1}	4.7×10^{-3}	14.1×10^{-3}	3.3×10^{-3}	1.4×10^{-3}
5	9.78×10^{-8}	2.05×10^{-8}	2.5	6.8	1.7	0.7
10	6.25×10^{-7}	1.19×10^{-7}	16.5	39.2	11.0	3.9
15	8.98×10^{-7}	1.60×10^{-7}	23.8	52.8	16.5	5.3
20	9.18×10^{-7}	1.56×10^{-7}	24.3	51.4	16.9	5.1
50	4.46×10^{-7}	6.33×10^{-8}	11.8	20.9	8.2	2.1
100	2.23×10^{-8}	2.23×10^{-8}	4.7	7.4	3.3	0.7
			Daytime Average			
0.37	9.45×10^{-21}	2.49×10^{-21}	2.5×10^{-13}	8.2×10^{-13}	1.7×10^{-13}	8.2×10^{-14}
0.89	1.65×10^{-17}	4.04×10^{-18}	4.3×10^{-7}	1.3×10^{-9}	3.0×10^{-7}	1.3×10^{-10}
5	3.83×10^{-11}	8.92×10^{-10}	1.0×10^{-3}	2.9×10^{-1}	6.9×10^{-4}	2.9×10^{-2}
10	3.00×10^{-9}	6.78×10^{-10}	8.0×10^{-2}	2.2×10^{-1}	5.6×10^{-2}	2.2×10^{-2}
15	8.98×10^{-9}	1.85×10^{-9}	2.4×10^{-1}	6.1×10^{-1}	1.7×10^{-1}	6.1×10^{-2}
20	1.23×10^{-8}	2.28×10^{-9}	3.3×10^{-1}	7.5×10^{-1}	2.3×10^{-1}	7.5×10^{-2}
50	8.28×10^{-9}	1.29×10^{-9}	2.2×10^{-1}	4.3×10^{-1}	1.5×10^{-1}	4.3×10^{-2}
100	3.18×10^{-9}	4.56×10^{-10}	8.4×10^{-2}	1.5×10^{-1}	5.8×10^{-2}	1.5×10^{-2}

factor of $t^{-0.2}$. If only one-half the total activity is in the cloud, the resulting equation is:

$$\text{Dose (in rem)} = 1.086 \times 10^4 \left[\frac{\text{1/2 Power (in Mw)}}{\text{Wind speed (in mps)}}\right] t^{-0.2} I_2.$$

The values of I_2 can be found from a nomogram given by Waterfield.[17] No results for daytime are given since the parameters for this case are beyond the scales of the nomogram. It is estimated that the daytime results will be less than those for the nighttime by a factor of about 15.

TABLE 6.9. EXTERNAL DOSE FROM HOT CLOUD AT NIGHT

Distance (miles)	Dose (rem)	Distance (miles)	Dose (rem)
0.37	1.81	15	3.43
0.89	1.87	20	3.12
5	2.44	50	1.18
10	3.51	100	0.51

Rainout from the Hot Cloud

One of the major hazards from any reactor disaster would be the release of the fission products into a cloud through which a rain shower would fall. In this case the scrubbing efficiency of the rain would depend on the rate of rainfall, the size of the droplets, and the relative velocity of the droplets with respect to the cloud. These may all be maximized for the purposes of this report if the total amount of activity in the cloud is considered to be deposited directly on the ground under the cloud by a hard shower. This spreading of the contained activity over the area of the cloud would result in decreasing contamination with distance as the cloud spread and as the activity decayed. To give the maximum hazard, the deposition at the ground in the center of the cloud was calculated and the effect of the Gaussian distribution was ignored. The ground exposure following a rainout of the hot cloud is given in Table 6.10. The dose rate in rem/hr was obtained by multiplying the surface contamination in curies/m^2 by 10.

TABLE 6.10. GROUND EXPOSURE FOLLOWING RAINOUT OF THE HOT CLOUD

Distance from Source (miles)	External Dose (rem/hr)	
	Average Daytime	Average Nighttime
0.37	1.7×10^4	4.6×10^5
0.89	1.4×10^4	3.5×10^5
5	4.2×10^3	1.0×10^5
10	1.8×10^3	4.5×10^4
15	1.0×10^3	2.6×10^4
20	6.5×10^2	1.8×10^4
50	1.5×10^2	4.5×10^3
100	4.7×10^1	1.6×10^3

The probability of the occurrence of rain at the time of an accident may be obtained from the following data:

Month	Hours of Rain (%)
January	11.4
February	7.5
March	8.6
April	4.4
May	6.1
June	2.5
July	3.2
August	2.2
September	3.1
October	2.0
November	3.8
December	10.9
Annual Average	6.5

Based on two years data, the results represent the percentages of all hours during which 0.01 in. or more of rain occurred.

The area covered by any rainout may be determined from the following equation[17] for the radius of the cloud, if it is assumed that the cloud is projected on the earth's surface:

$$r = \left[2.303\, c^2\, (x_o)^{2-n}\right]^{1/2}$$

where

r = radius in meters,
c = diffusion parameter,
n = stability coefficient,
x_o = corrected distance.

The meteorological parameters may be obtained from the preceding sections. The above equation defines the edge of a cloud as being where the activity is 10% that of the peak. An average dose over the affected area may be obtained by assuming the total activity in the cloud to be uniformly distributed over the area.

Exposure from a Cold Cloud

If it is assumed that both the pressure shell and the reactor cell are ruptured so that the fission activity could become airborne and, furthermore, that this is effected with no heat so that the subsequent expulsion of this activity by the building ventilators would constitute an instantaneous ground source, extremely high overdoses would result. The data are given in Table 6.11. The form of the table and the method for obtaining the various data are the same as for the hot cloud case presented in the previous section. Although it is believed to be a virtual certainty that the reactor cell would not be violated in any conceivable accident, if it were violated it is almost equally certain that it would be accompanied by the release of large quantities of heat (Appendix B) so that the exposure calculations pertinent to the hot cloud would apply.

Beryllium Hazard

The ART will contain 3000 lb of beryllium in the island and reflector, and therefore it presents a potential beryllium hazard. The peak concentration that should be permitted for a single exposure to beryllium[18] is 2.5×10^{-5} g/m^3. However, the beryllium metal cannot become airborne in any significant quantity. Although its melting point (2400°F) is slightly above that of the vaporization point of the NaF-ZrF_4-UF_4 fuel mixture, its vapor pressure is very low, and any molten beryllium would alloy with the Inconel. If there were sufficient oxygen present, some BeO would be formed, but the resultant hazard would be much less than that with the ARE wherein the beryllium was all present as the oxide.

18. J. H. Sterner and M. Eisenbud, "Epidemiology of Beryllium Intoxication", Arch. Ind. Hyg. Occupational Med. 4, 146 (1951).

TABLE 6.11. TOTAL INTEGRATED INTERNAL DOSES (TID) FROM COLD CLOUD

(1) Distance (miles)	(2) Integrated Dose per Curie, no decay (curie·sec/m^3)	(3) Integrated Dose per Curie, $t^{-0.2}$ decay (curie·sec/m^3)	(4) TID (Burnett) (curie·sec/m^3)	(5) TID (Marley) (curie·sec/m^3)	(6) Overdose (units of 25 rem) (Burnett)	(7) Overdose (units of 25 rem) (Marley)
			Nighttime Average			
0.37	6.2×10^{-4}	1.95×10^{-4}	16,000	64,000	11,000	6,400
0.89	2.4×10^{-4}	6.2×10^{-5}	6,400	20,000	4,400	2,000
5	2.2×10^{-5}	3.9×10^{-6}	580	1,300	400	130
10	7.6×10^{-6}	1.2×10^{-6}	200	400	140	40
15	4.1×10^{-6}	5.8×10^{-7}	110	190	77	19
20	2.6×10^{-6}	3.5×10^{-7}	69	120	48	12
50	6.0×10^{-7}	6.9×10^{-8}	16	23	11	2.3
100	2.0×10^{-7}	2.0×10^{-8}	5.3	6.6	3.7	0.66
			Daytime Average			
0.37	5.2×10^{-5}	1.7×10^{-5}	1,400	5,600	970	560
0.89	1.3×10^{-5}	3.5×10^{-6}	340	1,200	240	120
5	6.6×10^{-7}	1.3×10^{-7}	18	43	13	4.3
10	2.0×10^{-7}	3.4×10^{-8}	5.3	11	3.7	1.1
15	9.6×10^{-8}	1.5×10^{-8}	2.5	4.9	1.7	0.49
20	5.8×10^{-8}	8.6×10^{-9}	1.5	2.8	1.0	0.28
50	1.2×10^{-8}	1.4×10^{-9}	0.32	0.46	0.22	0.05
100	3.4×10^{-9}	3.7×10^{-10}	0.09	0.12	0.06	0.01

Appendix A

CHARACTERISTICS OF SITE

The physical characteristics of the site, as well as the population distribution and industrial complex of the surrounding area, are important considerations in the selection of any reactor site. The site characteristics of the proposed location of the ART in Oak Ridge are common to all other reactor proposals for the Oak Ridge area and therefore it was not deemed necessary to present a detailed analysis of these considerations in this report. However, brief summaries of the meteorology, climatology, geology, hydrology, and seismology of the site, as well as the industrial and population distribution in the surrounding areas, are given, with appropriate references to the detailed source material.

Meteorology and Climatology

A study of the meteorology and climatology at the proposed ART site was made in conjunction with the hazards analyses of the Aircraft Reactor Experiment and is reported in the ARE Hazards Summary Report.[1] A detailed report, including not only data from the ARE site but data from the entire Oak Ridge area, has since been issued by the U. S. Weather Bureau.[2] A summary of the wind direction, temperature gradients, and rainfall data, as presented in these reports, is included here.

The valleys in the vicinity of the ARE are oriented northeast-southwest, in roughly the same orientation as the broad valley between the Cumberland Plateau and the Smoky Mountains. As might be expected, considerable channeling of the winds results from this orientation. The direction of the prevailing winds is upvalley from southwest and west-southwest, with a secondary mass of downvalley winds from northeast and east-northeast. Wind speed is, usually, quite low, averaging less than 4 mph. In general, during nighttime or in stable conditions, the winds tend to be northeast and east-northeast and rather low in the valley, regardless of the gradient wind. Very strong winds aloft, however, will control the velocities and direction of the valley winds, reversing them or producing calms when opposing local drainage. In a well-developed stable situation, however, a very light air movement will follow the valley as far downstream as the valley retains its structure. Air transport from the valley location will be governed by the local valley wind and the degree of coupling winds.

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1. J. H. Buck and Wm. B. Cottrell, Aircraft Reactor Experiment Hazards Summary Report, ORNL-1407 (Nov. 24, 1952).

2. R. L. Myers and J. Z. Holland, A Meteorological Survey of the Oak Ridge Area, ORO-99 (Nov. 1953).

Two special wind patterns are assumed to be of some significance: (1) from the 7500 Area northwest of Haw Ridge to X-10, and (2) from the 7500 Area west to White Oak Creek, then northwest through Haw Gap, and finally north to X-10. Studies show that the frequency of these wind patterns is 2.5% over the ridge and 0.4% through the gap.

Since the upper wind pattern at Knoxville seems almost identical with that for Oak Ridge, the longer period records from the Knoxville Area have been used for this study. The northeast-southwest axis of the valley between the Cumberland Plateau and the Smoky Mountains, as mentioned before, influences the wind distribution over the Tennesee Valley, up to 5000 ft. Above 5000 ft, this pattern gives way to the prevailing westerly winds usually observed at these latitudes. Consideration of the relationship between precipitation and winds shows that there is little correlation between wind direction and rain.

The lower layers of the atmosphere tend to be stable more frequently than unstable, with inversions occurring 56% of the time, annually. In general, the stability is much more pronounced in the deep layer of air 183 to 5000 ft than in the 183-ft layer above the ground.

Distribution of Population

The population distribution within 30 miles of the site of the ART is summarized in Tables A.1, A.2, and A.3. A 30-mile radius was used for the population study because the meteorological studies showed that under certain conditions following a disaster a significant fraction of the maximum radiation dose would be received at these distances. Table A.1 presents the total number of employees at the various plant sites within the AEC restricted area at Oak Ridge. Although practically all these employees work a five-day week, there is considerable variance at the different plants in the number on any one shift. Table A.2 lists the surrounding towns with a population of 500 or more. Table A.3 gives the rural population by counties for those parts of the counties within 0 to 10, 10 to 20, and 20 to 30 miles of the site of the ARE. The latter data were calculated by deducting the urban (communities of 500 or more) population and assuming that the remaining population is uniformly distributed. These data are therefore approximate and are intended to give only an order of magnitude. Fig. A.1 shows the surrounding counties and all towns therein with a population greater than 500.

Vital Industries and Installations

A list of vital industrial and defense installations within possible hazard radius (30 miles) of the site of the ART is given in Table A.4. Most of these installations are shown on Fig. A.1.

TABLE A.1 PERSONNEL WITHIN THE AEC RESTRICTED AREA*

Plant	Distance from ART (miles)	Direction	Total Number of Employees	% of Time Downwind Night	% of Time Downwind Day
Homogeneous Reactor Test	0.24	SW	20	7.7	11.9
Oak Ridge National Laboratory (X-10), including 7000 Area	0.6	NW	2,700	1.4	2.5
University of Tennessee Agriculture School Extension	6.0	NE	54	11.6	9.0
Tower Shielding Facility	1.75	S	15	5.5	11.9
Gaseous Diffusion Plants	4.9	WNW	7,300	0.5	1.1
Operating Personnel			6,400		
Construction Personnel			900		
Electromagnetic Plant (Y-12)	5.1	NNE	15,600	5.6	5.5
ORNL Personnel			700		
Operating Personnel			5,400		
Construction Personnel			9,500		

*The people involved in current construction work are given separately from operating personnel, since most of the construction work now going on will be completed by the time the ART is scheduled to operate.

TABLE A.2 POPULATION OF THE SURROUNDING TOWNS*

City or Town	Distance from ART (miles)	Distance	Population	% of Time Downwind Night	Day
Oak Ridge	7	NNE	34,000	5.6	5.5
Lenoir City	9	SSE	5,159	4.3	6.0
Oliver Springs	9	N by W	1,089	2.3	2.7
Martel	10	SE	500	1.4	2.8
Coalfield	10	NW	650	0.5	1.1
Windrock	10	N by W	550	2.3	2.7
Kingston	12	WSW	1,627	9.5	11.3
Harriman	13	W	6,389	2.2	3.7
South Harriman	13	W	2,761	2.2	3.7
Petros	14	NW by N	790	1.4	2.8
Fork Mountain	15	NNW	700	2.3	2.7
Emory Gap	15	W	500	2.2	3.7
Friendsville	15	SE	600	1.4	2.8
Clinton	16	NE	3,712	11.6	9.0
Powell	17	ENE	500	8.3	6.8
Lyons View	18	E	500	1.5	2.7
Briceville	19	NNE	885	5.6	5.5
Wartburg	20	NW by W	800	1.4	2.8
Stainville	20	N	500	9.5	6.1
Knoxville	18 to 25	E	124,183	1.5	2.7
Greenback	20	S by E	960	5.5	4.9
Rockwood	21	W by S	4,272	2.2	3.7
Inskip	21	E	685	1.5	2.7
Rockford	22	SE	950	1.4	2.8
Whittle Springs	22	ENE	675	8.3	6.8
Fountain City	22	ENE	11,500	8.3	6.8
Gobey	22	NW	513	0.5	1.1
Lake City	23	NNE	1,827	5.6	5.5
Norris	23	NNE	1,134	5.6	5.5
Sweetwater	23	SSW	4,119	8.4	12.7
Neuberts	27	ENE	600	8.3	6.8
John Sevier	27	E	752	1.5	2.7
Madisonville	27	S	1,487	5.5	11.9
Caryville	27	N by E	1,234	9.5	6.1
Sunbright	30	NW	600	0.5	1.1
Jacksboro	30	N by E	577	9.5	6.1
Niota	30	SSW	956	8.4	12.7

*Included are those towns within a 30-mile radius that had a population of 500 or more according to the 1950 census, as reported in the 1952 Edition of the Rand-McNally Commercial Atlas and Marketing Guide, 83rd Ed. (1952). The Oak Ridge figure, however, is a current estimate.

TABLE A.3 RURAL POPULATION IN THE SURROUNDING COUNTIES

County	Total Area (sq. mi.)	Rural Pop. Density[a] (No. people per sq. mi.)	AREA (sq. mi.)			POPULATION		
			Within 0- to 10-mi. Radius	Within 10- to 20-mi. Radius	Within 20- to 30-mi. Radius	Within 0- to 10-mi. Radius	Within 10- to 20-mi. Radius	Within 20- to 30-mi. Radius
Anderson	338	62[b]	5[c]	175[c]	108	310	10,850	6,700
Blount	584	67	0	102	250	0	6,840	25,000
Campbell	447	48	0	0	130	0	0	6,240
Cumberland	679	22	0	0	80	0	0	1,760
Knox	517	178	55	140	208	9,780	24,900	37,000
Loudon	240	54	70	147	23	3,780	7,940	1,240
McMinn	435	39	0	0	69	0	0	2,690
Meigs	213	29	0	0	40	0	0	1,160
Monroe	665	26	0	35	212	0	910	5,510
Morgan	539	22	9	144	210	195	3,165	4,620
Rhea	335	33	0	0	52	0	0	1,720
Roane	379	50[b]	93[d]	185	59	4,650	9,250	2,950
Scott	549	26	0	6	111	0	155	2,890
Union	212	38	0	0	15	0	0	570
						18,715	63,910	100,050

(a) Includes all county population except communities with population of 500 or more.
(b) Does not include the Oak Ridge area.
(c) Does not include the Oak Ridge area in Anderson county.
(d) Does not include 42 square miles of Oak Ridge area in Roane county.

TABLE A.4 VITAL INDUSTRIAL AND DEFENSE INSTALLATIONS IN 30-MILE RADIUS

Industry or Installation	Distance from ART (miles)	Direction	% of Time Downward Night	% of Time Downward Day
Homogeneous Reactor Test	0.24	SW	7.7	11.9
ORNL, X-10 Site	0.60	NW	1.1	0.9
Tower Shielding Facility	1.75	S	5.5	11.9
Gaseous Diffusion Plants	4.9	WNW	0.6	1.4
Electromagnetic Plant (Y-12)	4.9	WNW	5.6	5.5
Fort Loudon Dam	10.0	SSE	4.3	6.0
Kingston Steam Plant (TVA)	11.0	W	2.2	3.7
Assorted Small Industries in Knoxville	20 to 26	E	1.5	2.7
Aluminum Company of America, Alcoa Plant	22.0	SE	1.4	2.8

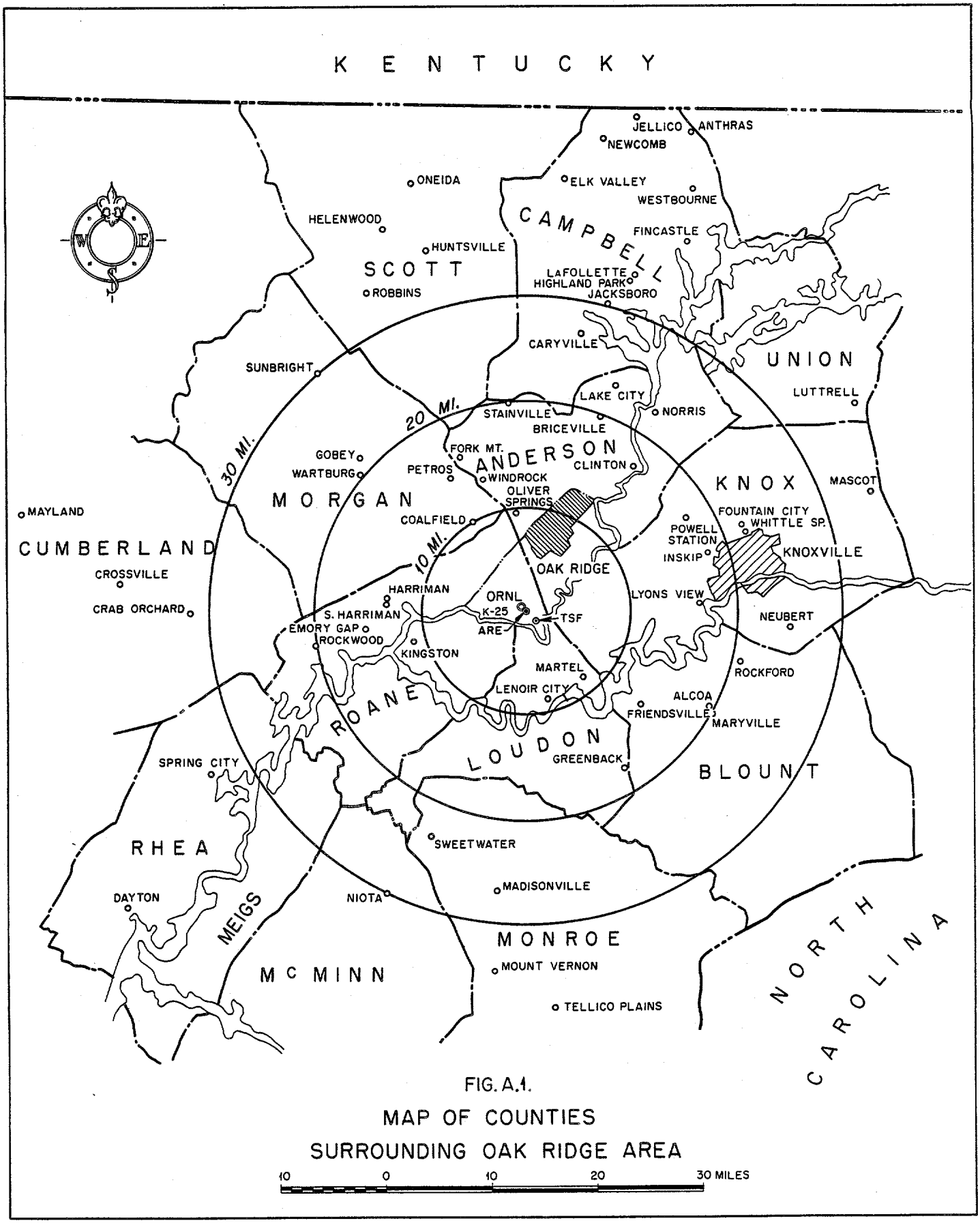

FIG. A.1.

MAP OF COUNTIES

SURROUNDING OAK RIDGE AREA

Geology and Hydrology of Site

Melton Valley, in which the ART would be situated, is underlaid by the Conasauga shale of the Middle and Upper Cambrian Age. The more resistant rock layers of the Rome formation, steeply inclined toward the southwest, are responsible for Haw Ridge, which is immediately northwest of the ART site. These layers dip beneath the shales of the conasauga group in Melton Valley. The shale layers in the area are in keeping with the general structure of the surrounding area as reported in a recent survey.[3] Thin layers and lenses of limestone are common but are generally irregular in distribution. However, there are no persistent limestone beds in the area and, consequently, no underground solution channels or caverns to permit rapid and free discharge of water underground.

Observations in test wells in soil comparable to that of the ART site show that the Conasauga shale, although relatively impermeable, is capable of transmitting small amounts of ground water at a rate of a few feet per week. Furthermore, all of the active isotopes, except for ruthenium, apparently become fixed in the immediate vicinity of the point of entry into the soil. It may be concluded that such ground water flow as may exist in the soil surrounding the ART will be small and slow (a few feet per week) and that such flow will reduce the level of the activity of mixed nonvolatile fission products more than 90%.

Aside from rainout of airborne wastes, the only conceivable sources of ground water flow are (1) the normal discharge of process water and (2) the leakage of water used to flood the offgas pipe pit. The fuel does not react with water and in its presence would solidify. Process water will not come in contact with the fuel, except as a result of a disaster. It is therefore improbable that the process water would contain appreciable radioactivity. The water at the point of discharge could create a surface stream. From this point, 100 ft southwest of the ART building, the stream would flow south and join Melton Creek about 1/2 mile above White Oak Creek. The offgas pipe trench will permit decay of the activity induced in the water by the off gases. Leakage, if any, should be small and subsurface.

In view of the lack of serious water contamination, as well as the extremely favorable geological environment, there appears to be even less ground water hazard associated with the operation of the ART than with the ARE, for which the hazard was negligible.

Seismology of Area

Information on the frequency and severity of earthquakes in East Tennessee has been obtained both from Lynch[4] of the Fordham University Physics Department and from Moneymaker[5] of the Tennessee Valley Authority. Both

3. P. B. Stockdale, Geologic Conditions at the Oak Ridge National Laboratory (X-10) Area Relevant to the Disposal of Radioactive Waste, ORO-58 (Aug. 1, 1951).

4. Letter from J. Lynch to M. Mann, Nov. 3, 1948, quoted in A Report on the Safety Aspects of the Homogeneous Reactor Experiment, ORNL-731

5. B. C. Moneymaker, a private communication to W. B. Cottrell, Oct.27, 1952.

sources indicated that such shocks as occasionally occur in the region are quite common in the world and do not indicate undue seismic activity. Consequently, earthquakes should be of little concern in connection with the ART.

The TVA records show that the Appalachian Valley from Chattanooga to Virginia has an average of only one or two earthquakes a year. Furthermore, the maximum intensity of any of these shocks is 5 on the Woods-Neuman scale. This intensity is barely noticeable by ambulatory as well as stationary individuals. For any one location, such as Oak Ridge, the expectancy of an earthquake would be one in every few years.

The Fordham University records indicate even lower quake frequency; however, the severity of the observed quakes is the same. Lynch further concluded that "it is highly improbable that a major shock will be felt in the area (Tennessee) for several thousand years to come."

Appendix B

HEAT RELEASED IN CHEMICAL REACTIONS AND RESULTING TEMPERATURES AND PRESSURES

The chemical reactions which are potential sources of large amounts of energy have been considered. Calculations have been made to find the consequences of 1000 lb of sodium and NaK reacting with air or with water, and the reaction of the zirconium-base fuel with sodium. The reactions of the zirconium-base fuel with NaK and of the lithium-base fuel with either sodium or NaK were found to be less severe. Hence calculations on these reactions have been omitted.

Computations that give the temperature and pressure resulting from the reaction of the sodium and the NaK with the 34,000 lb of shield water are also presented. This case is then considered when the reaction is accompanied by the dispersion of hydrogen gas and the NaOH, as dust, through the atmosphere of the cell. Finally a treatment is given of the sodium and NaK reacting with water accompanied by burning of the hydrogen formed in air.

The thermodynamic data basic to all the following calculations are given in Table B.1.

Table B.1

Basic Thermodynamic Data

Material	Specific Heat (cal/g-mole °C)	Material	Heat of Formation (kcal/g-mole)
KOH	19.6	KOH	-102.02
NaOH	19.6	NaOH	-101.96
Na_2O	16.3	Na_2O	- 99.45
NaF	11.0	K_2F	- 86.4
KF	11.7	NaF	-136.0
K	7.0	KF	-134.5
Na	7.0	ZrF_4	-445.0
Zr	6.36	UF_4	-443.0
$NaF-ZrF_4-UF_4$*	34.2	H_2O	- 68.32
H_2	7.41 (18 to 1700°C)		
N_2	8.27 (18 to 3000°C)		

Density of N_2 = 0.0785 lb/ft^3
Density of H_2 = 0.00561 lb/ft^3
Density of air = 0.08071 lb/ft^3

* 50-46-4 mole %

Reaction of 70 Pounds of Sodium with Air

$$4\ Na + O_2 \longrightarrow 2\ Na_2O$$

Assume the sodium temperature to be 815°C and the air temperature to be 18°C and take all temperatures with respect to 18°C.

Enthalpy of Na = (7.0) (815-18) = 5590 cal/mole Na
Enthalpy of Na_2O = (16.3) ΔT cal/mole Na_2O
Enthalpy of N_2 = (8.27) ΔT cal/mole N_2
Heat of Formation of Na_2O = -99.45 kcal/mole Na_2O

Air contains 4.30 moles of N_2 for each mole of O_2.
Consider no excess of air for a stoichiometric combination.

$$4\ Na + O_2 + 4.30\ N_2 \longrightarrow 2Na_2O + 4.30\ N_2$$

$$4\ (5590) = 4.30\ (8.27)\Delta T + 2(16.3\Delta T - 99{,}450)$$

$$\Delta T = 3250^\circ C$$

$$T = 3268^\circ C$$

70 lb of Na = 1380 g-moles of Na

Heat of formation of Na_2O = -99.45 kcal for 2 moles of Na

Heat release = $99.45 \times \frac{1380}{2} = 6.86 \times 10^4$ kcal

$= 2.72 \times 10^5$ Btu

Reaction of 70 Pounds of Sodium with Water

When sodium and water are combined in stoichiometric proportions and the total heat of the reaction is absorbed by the products of the reaction, the reaction products will be heated to a temperature which is the maximum temperature for the reaction.

$$2\ Na + 2H_2O \longrightarrow 2NaOH + H_2 \uparrow$$

The temperature of the sodium in the reactor will be 815°C, and the temperature of the shield water will be 60°C. Consider temperatures with respect to a base temperature of 18°C:

Enthalpy of Na = (7.0) (815-18) = 5,600 cal/mole Na
Enthalpy of H_2O = (1) (60-18) = 42 cal/mole H_2O
Enthalpy of NaOH = (19.6) ΔT cal/mole NaOH
Enthalpy of H_2 = (7.41) ΔT cal/mole H_2
Heat of formation of H_2O = -68.32 kcal/mole H_2O
Heat of formation of NaOH = -101.96 kcal/mole NaOH

$$2Na + 2H_2O \longrightarrow 2NaOH + H_2\uparrow$$

$$2\,(5{,}600) + 2\,(42\text{-}68{,}320) = 2(19.6\,\Delta T - 101{,}960) + 7.41\,\Delta T$$

$$\Delta T = 1680^{\circ}C$$

$$T = 1698^{\circ}C$$

To calculate the heat generated by the reaction of 70 lb of sodium with water, the number of moles of sodium must be multiplied by the net heat of formation per mole.

70 lb of Na $= \frac{70}{23}\,454 = 1{,}380$ g-mole Na

Net heat of formation $= -101.96-(-68.32) = -33.64$ kcal/mole Na

Heat release $= 33.64\,(1{,}380) = 4.56 \times 10^4$ kcal

$= 1.84 \times 10^5$ Btu

Reaction of 930 Pounds of NaK with Air

$$4\,Na + O_2 \longrightarrow 2Na_2O$$

$$4\,K + O_2 \longrightarrow 2K_2O$$

The NaK temperature in the reactor will be 815°C.
Consider the air to be at 18°C and take all temperatures with respect to 18°C.

Enthalpy of Na $= (7.0)\,(185\text{-}18) = 5590$ cal/mole Na
Enthalpy of K $= (7.0)\,(815\text{-}18) = 5590$ cal/mole K
Enthalpy of Na_2O $= (16.3)\,\Delta T$ cal/mole Na_2O
Enthalpy of K_2O $= (16.3)\,\Delta T$ cal/mole K_2O
Enthalpy of N_2 $= (8.27)\,\Delta T$ cal/mole N_2

Heat of formation of Na_2O $= -99.45$ kcal/mole Na_2O

Heat of formation of K_2O $= -86.4$ kcal/mole Na_2O

Since 930 lb of NaK consists of 410 lb of sodium and 520 lb of potassium, the heat balance will have to be adjusted according to the fractions of Na_2O and K_2O produced.

Air contains 4.30 moles of N_2 for every mole of O_2. Consider no excess air for a stoichiometric combination.

$$1.6\,Na + 2.4\,K + O_2 + 4.30\,N_2 \longrightarrow 4.30\,N_2 + 0.8\,Na_2O + 1.2\,K_2O$$

$$(1.6 + 2.4)\,5590 = 4.30\,(8.27)\,\Delta T + 0.8\,(16.3\,\Delta T - 99{,}450)$$

$$+ 1.2\,(16.3\,\Delta T - 86{,}400)$$

$$\Delta T = 3001^{\circ}C$$

$$T = 3019^{\circ}C$$

$$410 \text{ lb of Na} = \frac{410}{23} 454 = 8090 \text{ g-moles Na}$$

$$520 \text{ lb of K} = \frac{520}{39.1} 454 = 6030 \text{ g-moles K}$$

Heat of formation of Na_2O = -99.45 kcal/2 moles Na
Heat of formation of K_2O = -86.4 kcal/2 moles K
Heat release = 99.45 (8090/2) + 86.4 (6030/2) = 6.62×10^5 kcal
= 2.63×10^6 Btu

Reaction of 930 Pounds of NaK with Water

In 930 lb of NaK there are 410 lb of sodium and 520 lb of potassium.

Heat of formation of H_2O = -68.32 kcal/g-mole
Heat of formation of NaOH = -101.96 kcal/g-mole
Heat of formation of KOH = -102.02 kcal/g-mole

Net heat of formation for the Na-H_2O reaction = -101.96-(-68.32)
= -33.64 kcal/g-mole

Net heat of formation for the K-H_2O reaction = -102.02-(-68.32)
= -33.70 kcal/g-mole

$$410 \text{ lb of Na} = \frac{410}{23} 454 = 8080 \text{ g-moles Na}$$

$$520 \text{ lb of K} = \frac{520}{39.1} 454 = 6020 \text{ g-moles K}$$

Heat release = 33.64 (8080) + 33.70 (6060) = 4.76×10^5 kcal
= 1.89×10^6 Btu

Reaction of 1200 Pounds of Fuel with Sodium

$$4 \text{ Na} + UF_4 \longrightarrow 4NaF + U$$

$$4 \text{ Na} + ZrF_4 \longrightarrow 4 \text{ NaF} + Zr$$

Consider the fuel to be NaF-ZrF_4-UF_4 (50-46-4 mole %).

1200 lb of fuel contains 228 lb NaF
837 lb ZrF_4
135 lb UF_4

The fuel temperature in the reactor will be 900°C; the NaK temperature will be 815°C. Consider all temperatures with respect to a base temperature of 18°C.

Enthalpy of Na = 7.0 (815-18) = 5,590 cal/mole Na
Enthalpy of fuel 30 = 34.2 (900-18) = 30,200 cal/mole fuel 30
Enthalpy of NaF = 11.0 ΔT cal/mole NaF
Enthalpy of Zr = 6.36 ΔT cal/mole Zr
Enthalpy of U = 6.7 ΔT cal/mole U

Heat of formation of UF_4 = -443 kcal/mole UF_4

Heat of formation of ZrF_4 = -445 kcal/mole ZrF_4

Heat of formation of NaF = -136 kcal/mole NaF

$$2Na + (0.04\ UF_4 + 0.46\ ZrF_4 + 0.50\ NaF) \longrightarrow 2.5\ NaF + 0.04\ U + 0.46\ Zr$$

$$2\ (5{,}590) + 30{,}200 + 0.04\ (-443) + 0.46\ (-445) + 0.50\ (-136)$$

$$= 2.5\ (-136) + 2.5\ (11.0)\Delta T + 0.04\ (6.7)\Delta T + 0.46\ (6.36)\Delta T$$

$$\Delta T = 2960^{\circ}C$$

$$T = 2978^{\circ}C$$

$$837 \text{ lb of } ZrF_4 = \frac{837}{167.2}\ 454 = 2280 \text{ g-mole } ZrF_4$$

$$135 \text{ lb of } UF_4 = \frac{135}{311}\ 454 = 197 \text{ g-mole } UF_4$$

Net heat of formation for the ZrF_4-Na reaction = 4 (136) - 445 = 99 kcal/mole ZrF_4

Net heat of formation for the UF_4-Na reaction = 4 (136) - 443 = 101 kcal/mole UF_4

Heat release = 99 (2280) + 101 (197) = 2.46×10^5 kcal
= 9.77×10^5 Btu

Reaction of Sodium and NaK with Shield Water in Nitrogen

Heat Absorbed in Water. Assume that the reactor cell contains 12,000 ft^3 of N_2 at a pressure of 14.7 psia and a temperature of 80°F.

Heat produced by Na and NaK-H_2O reaction	=	2.07×10^6 Btu
Heat from a nuclear excursion	=	0.30×10^6 Btu
Total reaction heat added to the cell	=	2.37×10^6 Btu

Enthalpy of 480 lb Na at 1500°F	=	2.19×10^5 Btu
Enthalpy of 520 lb K at 1500°F	=	1.39×10^5 Btu
Enthalpy of 34,000 lb H_2O at 140°F	=	2.64×10^5 Btu
Enthalpy of 942 lb N_2 at 80°F	=	0.19×10^5 Btu
Total enthalpy	=	6.41×10^5 Btu

Principal quantities present after the reaction:

33.6 lb moles of N_2	C_p = 6.97 Btu/lb·mole·°F	(60 to 100°F)
17.1 lb moles of H_2	C_p = 6.91 Btu/lb·mole·°F	(60 to 100°F)
20.9 lb moles of NaOH	C_p = 19.6 Btu/lb·mole·°F	
13.3 lb moles of KOH	C_p = 19.6 Btu/lb·mole·°F	
1,856 lb moles of H_2O	$C_{p(vap)}$ = 8.09 Btu/lb·mole·°F	(60 to 100°F)

By assuming all heat goes into the reaction products, N_2, and shield water, three equations to relate final temperature, pressure, moles of water vapor, and total moles of vapor within the reactor cell can be written. The fourth equation, necessary for a solution, takes the form of a table relating the vapor pressure and temperature of water. Heat from the reaction must be equal to the heat absorbed by the reaction products and atmosphere.

$$\Sigma Q = \Sigma m C_p \Delta T$$

$$(2.37 + 0.64)\, 10^6 = \left[33.6\,(6.97) + 17.1\,(6.91) + 34.2\,(19.6) + N_v\,(8.09)\right]$$

$$(T - 460) + (1856 - N_v)\, 18\, (T - 460) + 17{,}510\, N_v$$

The final temperature, pressure, and number of moles of water vapor can be related by the ideal gas law.

$$PV = NRT$$

$$P = \frac{(50.7 + N_v)\, 10.7)}{12{,}000}\, T$$

The partial pressure of water vapor is directly proportional to its molar percentage.

$$\frac{P_v}{P} = \frac{N_v}{N}$$

$$P = (50.7 + N_v)\, \frac{P_v}{N_v}$$

Simultaneous solution of the three equations gives

T = 546°R = 86°F

P_v = 0.616 psia

N_v = 1.26 moles water vapor

P = 25.4 psia = 10.7 psig .

Heat Absorbed in Nitrogen. The maximum heat added to the cell atmosphere can be taken as 2.37×10^6 from the Na and NaK-water reaction plus 0.36×10^6 Btu from the sensible heat of the Na and NaK and 0.3×10^6 Btu from a severe nuclear accident. Equating the total to the products of the weights, specific heats, and temperature rise of the material, and assuming that the heat is distributed throughout the 12,230 ft^3 reactor cell by the H_2 and NaOH which is formed, we obtain

$$3 \times 10^6 = \left[(\text{moles of NaOH and KOH})\, C_p + (\text{moles of gas})\, C_p\right] \Delta T$$

$$3 \times 10^6 = \left[(34.2)\, 19.6 + (33.6)\, 8.44 + (17.1)\, 7.89\right] \Delta T$$

$$\Delta T = 2712^\circ F$$

Initial temperature of N_2 = $80^\circ F$

Initial pressure of N_2 = 14.7 psia

Initial number of moles of N_2 = 33.6

Final temperature of gasses = $2792^\circ F$

Final number of moles of gas = 50.7

$$P = 14.7 \frac{50.7}{33.6} \frac{3252}{540} = 133 \text{ psia} = 118 \text{ psig}$$

If a 1-in.-thick wall for the cylindrical section of the 24-ft-dia reactor cell, a peak pressure of 133 psia inside the tank, and a minimum head of 12 ft of water outside the tank are assumed, a pressure differential across the tank wall of 113 psi is obtained. The tangential stress in the wall then becomes:

$$S_t = P \frac{r}{t} = 113 \frac{144}{1} = 16{,}200 \text{ psi.}$$

The axial stress is:

$$S_a = P \frac{r^2 \pi}{2\pi r} = 113 \times 72 = 8{,}136 \text{ psi.}$$

The combined stress then becomes:

$$S_c = \sqrt{S_a^2 + S_t^2} = 18{,}000 \text{ psi.}$$

The stress for rupture in the steel tank wall will be about 60,000 psi, hence a pressure nearly four times the above value for an extreme accident would be required to rupture the tank.

It should be noted that the stress in the hemispherical ends of the tank will be the same as in the cylindrical section, even though their thickness is half as great. Care must, of course, be used in the detail design of the joints to avoid stress concentrations.

Reaction of Sodium and NaK with Shield Water in Air

A reactor cell filled with air and containing stoichiometric proportions of Na, NaK, and H_2O is assumed. It is further assumed that the H_2 generated by the Na and NaK-H_2O reactions burns in the air as quickly as it is generated.

17.1 lb-moles of H_2 are available for the reaction; but 12,000 ft^3 of air contain only 6.35 lb-moles O_2.

Heat of formation of H_2O = -68.32 kcal/g-mole H_2O

12.7 lb-moles = 5760 g-moles of H_2

Heat release = (5760) 68.32 = 3.94×10^5 kcal
= 1.56×10^6 Btu

Heat produced by Na and NaK-H_2O reaction	= 2.07×10^6 Btu
Sensible heat of the Na and NaK	= 0.36×10^6
Heat from a nuclear excursion	= 0.30×10^6
Heat produced by H_2-O_2 reaction	= 1.56×10^6
Total reaction heat added to the cell	= 4.29×10^6 Btu

Principal quantities present after the reaction:

3.4 lb moles of H_2	C_p = 8.73 Btu/lb·mole·°F	(60 to 4700°F)
27.3 lb moles of N_2	C_p = 8.84 Btu/lb·mole·°F	(60 to 4700°F)
20.9 lb moles of NaOH	C_p = 19.6 Btu/lb·mole·°F	
13.3 lb moles of KOH	C_p = 19.6 Btu/lb·mole·°F	
12.7 lb moles of H_2O	C_p = 13.0 Btu/lb·mole·°F	(60 to 4700°F)

Heat balance:

$$4.29 \times 10^6 = \left[3.4\ (8.73) + 27.3\ (8.84) + 34.2\ (19.6)\right] \Delta T$$

$$17{,}510\ (12.7) + 12.7\ (212) + 12.7\ (13.0)\ (\Delta T - 212)$$

$$\Delta T = 3715^\circ F$$

The pressure in the reactor cell can be found by use of the ideal gas law.

Initial temperature of air = 80°F

Initial pressure of air = 14.7 psia

Initial number of moles of gas = 27.3

Final temperature of atmosphere = 3715°F

Final number of moles of gas = 43.3

$$P = 14.7 \frac{43.3}{27.3} \frac{4175}{540} = 181 \text{ psia}$$

$$= 166 \text{ psig}$$

Appendix C

METALLURGY AND CHEMISTRY

The probability and consequences of a leak in the ART, in view of the corrosion and the chemical reactions that would occur, will be comparable with those for the ARE. However, the ART will employ thinner metal walls and will operate at a higher power, and therefore the results of a leak have been re-examined in considerable detail. In the re-examination the known corrosion mechanisms for the $NaF-ZrF_4-UF_4$ fuel system in Inconel have been taken into consideration. The corrosion penetration by the $NaF-LiF-KF-UF_4$ fuel system appears to be comparable, but less is known of the actual corrosion behavior. The various chemical reactions between sodium or NaK and the ART fuel which would result if a leak occurred are known and have also been taken into account.

Corrosion of Inconel by the Fluoride Fuel

An extensive study of the corrosion of Inconel by molten fluorides (especially $NaF-ZrF_4-UF_4$ mixtures) has been conducted during the past several years. Most of the tests have been made with Inconel thermal-convection loops operated with a top temperature of 1500°F and a 200°F temperature gradient which induces flow rates of 2 to 6 fpm in the fluoride mixture. The effect of time, temperature, purity, ratio of surface area to volume, and concentration of uranium have been investigated. In general, data from the thermal-convection loops have been substantiated by the data from the few high-velocity, high-thermal-gradient loops that have been operated.

Inconel, a solid solution alloy of Ni, Cr, and Fe, corrodes through preferential leaching of chromium, the least noble of the constituent metals. The corrosion manifests itself in the formation of voids in the metal; these voids are subsurface, are not interconnected, and are, in general, not localized at grain boundaries. The diffusion gradient set up by the selective removal of chromium at the Inconel-fused salt interface produces a "concentration" of lattice vacancies higher than the equilibrium value in the solid solution structure. These lattice vacancies tend to coalesce to form macroscopic voids at disregistries in the lattice (the Kŭkendall effect[1]).

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1. See, for example, F. Seitz, "On the Porosity Observed in the Kŭkendall Effect", Acta Metallurgia 1, 355 (1953).

The selective leaching of chromium occurs, not because of physical solubility of chromium metal in molten fluorides, but by chemical reaction of this metal with oxidizing agents present in the melt or on the original metal surface. Accordingly, corrosion of Inconel by the fused salt is strongly dependent on the concentration of these reducible compounds; their dependence is emphasized as the ratio of fuel volume to metal surface area is increased.

Typical impurities react and produce corrosion by the following reactions:

$$2HF + Cr^{\circ} \longrightarrow CrF_2 + H_2$$
$$NiF_2 + Cr^{\circ} \longrightarrow CrF_2 + Ni^{\circ}$$
$$FeF_2 + Cr^{\circ} \longrightarrow CrF_2 + Fe^{\circ}$$
$$2FeF_3 + 3Cr^{\circ} \longrightarrow 3CrF_2 + 2Fe^{\circ}$$
$$2CrF_3 + Cr^{\circ} \longrightarrow 3CrF_2$$

Oxide films on the metal walls react with the fuel constituents (ZrF_4 or UF_4) to yield structural metal fluorides:

$$2NiO + ZrF_3 \rightarrow 2NiF_2 + ZrO_2$$
$$2Fe_2O_3 + 3ZrF_4 \rightarrow 4FeF_3 + 3ZrO_2$$
$$2Cr_2O_3 + 3ZrF_4 \rightarrow 4CrF_3 + 3ZrO_2$$

These structural metal fluorides are then available for reaction with chromium, as shown above. It is, accordingly, necessary that the melt and the test metal be of high purity if undue corrosion is to be avoided. If these conditions are realized, and if UF_4 is the uranium compound used, the reaction

$$2UF_4 + Cr^{\circ} \rightleftharpoons CrF_2 + 2UF_3$$

becomes an important contribution to the corrosion reaction.

In the thermal-convection loops, reduction of impurities in the melt and equilibration of the metal surface with the UF_4 seems to require 200 to 250 hr and to produce void formation to a depth of 3 to 5 mils. If the flow rate were increased, these reactions should proceed more rapidly but should cause no greater corrosion. For a given concentration of impurities and UF_4 the depth of attack varies with the ratio of surface area to fuel volume. If equilibrium is established isothermally, corrosion should be reasonably uniform. However, if these reactions are conducted in a system with a temperature gradient, the hottest zone will show preferential attack.

The last reaction shown above,

$$2UF_4 + Cr^{\circ} \rightleftharpoons CrF_2 + 2UF_3,$$

is, in addition, responsible for some "mass transfer" of chromium from the hot metal walls to the cold regions of the system. Since the equilibrium constant for the reaction is temperature dependent, the reaction proceeds

slightly further to the right at 1500°F than at 1150°F; since the UF_3 and CrF_2 are soluble they move with the melt to the 1150°F region where a slight reversal of the reaction occurs and chromium metal is formed. This type of conversion is hardly apparent in thermal-convection loops operated for 500 hr. It appears, however, that in such systems corrosion by this mechanism may be expected to increase by about 4 mils per 1000 hr.

This mass transfer phenomenon should be sensitive to absolute temperature, temperature gradient, UF_4 concentration, flow velocity, ratio of area to fuel volume, and rate of chromium diffusion in the metal. The rate of mass transfer is known to increase with increases in the first three variables listed; the quantitative effect of the other variables is not as yet known with certainty.

The corrosion attack in the ART would be greatest in the hottest portion of the fuel circuit, and, after 1000 hr of operation, the attack should be about 15 to 18 mils unless the higher flow velocities have a much greater effect on the mass transfer than can be hypothesized. In all the tests to date, radiation has not shown an increase in the corrosion of Inconel in the fused fluorides; however, this has not been studied in a flowing system with a temperature gradient. The possibility of plugging part of the heat exchanger circuit with chromium is considered to be extremely slight, because the amount of chromium that will be transferred during 1000 hr of operation will be very small. Experiments with high-velocities and high temperature gradients are under way which are expected to confirm these conclusions.

The use of a mixture of UF_3 and UF_4 in the fuel will greatly decrease the corrosion of Inconel by the fuel. For instance, after 500 hr of circulation of a UF_3-containing fused salt, attack of 1 to 2 mils is found rather than the 3 to 5 mils found with UF_4 fuels. The addition of UF_3 will always reduce the amount of mass transfer, since the mass transfer is proportional to $(UF_4/UF_3)^2$ in the region of interest.

Tests have also been made to determine the effects of a temperature excursion on corrosion. In these 500-hr tests in the temperature range 1350 to 1650°F, the attack changed from small voids throughout the grain at 1350°F to large voids predominantly at the grain boundaries at 1650°F. The depth of attack was practically the same in all these tests. If the temperature excursions were as high as 1900°F, the carbide particles would go into solution and rapid grain growth would occur; however, the depth of attack would not be increased more than a factor of 2 to 3.

In summation, it may be said that the zirconium-base fluoride fuels and Inconel are compatible at ART operating temperatures for the 1000-hr expected life and that the attack will not seriously weaken the reactor structure. The amount of mass transfer of chromium to the cold leg will be so small that there will not be an increase in pressure drop or decrease in heat transfer performance. Prior to the operation of the ART, however, sufficient experiments will have been run in high-flow-velocity, high-temperature-gradient loops to permit a more realistic statement of the corrosion to be expected.

Mass Transfer in the Sodium-Inconel-Beryllium System

One of the most serious compatibility problems in the ART is that found in the moderator circuit where sodium will flow in direct contact with the beryllium and with Inconel. This will give rise to dissimilar metal transfer between beryllium and Inconel with sodium as the carrier. The tests made to date in pump loops and thermal-convection loops have shown that, in general, dissimilar metal transfer is quite troublesome at temperatures above 1300°F; but, if the operating temperature of the interface does not exceed 1250°F, the amount of transfer is quite small. The beryllium in contact with the fast flowing sodium is relatively undisturbed and no dendritic beryllium crystals are found in the cold portion of systems when the upper temperature limit is 1250°F or below.

When the sodium between the beryllium and an Inconel wall is stagnant, the beryllium is attacked to a depth of 1 to 3 mils. This is due to the alloying of the beryllium with the Inconel and the formation of the intermetallic compounds BeNi and $Be_{21}Ni_5$, which do not spall. However, one of the most important variables in dissimilar metal transfer is the spacing between the two metals, and it has been found that if the spacing is as great as 50 mils, the dissimilar metal transfer is greatly minimized. To keep the Inconel and beryllium from coming into close contact, spacers, which are compatible with the sodium, Inconel, and beryllium, will be used. Therefore, with a spacing of 50 mils or greater between the Inconel and the beryllium and with the temperature limitation of 1300°F or lower, dissimilar metal transfer will be quite small.

Chemical Interaction of Fluoride Fuel and Na or NaK

The chemical interactions of Na or NaK with molten fluorides of the type to be used in the ART are well understood. If the fuel system is a $NaF-ZrF_4-UF_4$ mixture and sodium is added to an excess of the fused salt, the following reactions will occur in sequence:

$$(1)\quad UF_4 + Na^\circ \longrightarrow NaF + UF_3$$
$$(2)\quad ZrF_4 + 4Na^\circ \longrightarrow 4NaF + Zr^\circ$$
$$(3)\quad UF_3 + 3Na^\circ \longrightarrow 3NaF + U^\circ$$

When an excess of the alkali metal is present, these reactions occur simultaneously, and the heavy metals are produced immediately. If the fuel system contains only alkali fluorides and UF_4, the reaction of the alkali metal with a large excess of molten salt proceeds through the following sequential reactions:

$$(1)\quad UF_4 + Na^\circ \longrightarrow NaF + UF_3$$
$$(4)\quad 4UF_3 \rightleftharpoons 3UF_4 + U^\circ$$

The latter reaction (4) becomes appreciable only after considerable UF_3 is present (perhaps 60% of the total U is trivalent), but the equilibrium is

quite temperature sensitive. The reaction is noticeably more complete at higher temperatures. Reaction of this fused salt mixture with an excess of the alkali metal immediately produces uranium metal by the reaction

$$(5)\quad UF_4 + 4Na^o \longrightarrow 4NaF + U^o.$$

The consequences of mixing ART fuel and coolant can be realistically appraised for several hypothetical situations. However, it is not possible to foresee all possible situations under which the mixing could take place as a result of an ART accident; there must, accordingly, be some uncertainty in any evaluation of the consequences of such an accident.

The fuel of the ART could contact molten, alkali metal if a leak developed in either of two systems. Development of a leak between the fuel system and the primary heat exchanger would introduce NaK into the fuel circuit or fuel into the NaK stream. Since the primary heat exchanger will involve about 3000 small, thin-walled tubes and the consequent tube to header welds, it would seem to be the most likely source of leaks. A leak between the moderator-coolant circuit and the fuel would let Na^o into the fuel or fuel into the coolant channels in the beryllium moderator. While in principle it is possible for the solid reaction products (neither NaF, U^o, or Zr^o are molten at 1500°F) to plug small leaks, plugging is likely to occur only if the fuel and the alkali metal meet in an essentially stagnant spot. In view of the flow rates involved, this does not seem especially likely.

Heats of reaction are quite high for the interactions which produce U^o and Zr^o. If it were possible, for example, to mix ZrF_4-bearing fuel (200 lb UF_4, 900 lb ZrF_4, and 300 lb NaF) instantaneously with a stoichiometric quantity of NaK (80% K^o) and have the reaction proceed adiabatically, the temperature rise would be about 1200°C; the fuel temperature would be about 3300°F, and the uranium (but not the zirconium) formed would be molten. If the fuel system contained only UF_4 with alkali fluorides and these reaction conditions prevailed, the temperature rise would be somewhat less than 500°F; the fuel temperature would be about 1800°F.

Reactions from a Heat Exchanger Leak. Small leaks at the heat exchanger are potentially hazardous but do not appear to be immediately dangerous regardless of direction of the leakage. If the leak were sufficiently slow to permit attainment of equilibrium conditions (and the very rapid turbulent flow of the fuel system should encourage this), the reaction product with either fuel type would be UF_3. Accordingly the first evidence of a slow leak into the fuel would be slight dilution of the fuel and a consequent decrease in reactivity both because of dilution and of poisoning by K^o.

If the fuel were a NaF-ZrF_4-UF_4 mixture, about 5 lb of NaK would be required to raise the UF_3 concentration to a level which would permit precipitation of UF_3 at the lowest fluid temperature. Whether the insoluble material would deposit on and adhere to the metal surfaces in headers and pumps from this rapidly flowing stream is questionable; certainly, its

deposition would be far from complete. However, deposition on these regions would decrease reactivity in the reactor core. If the leak continued the solution would be supersaturated with UF_3 at increasing temperatures and, accordingly, over wider areas in the reactor system. As this occurred the loss of uranium from the system by deposition on the container metal should increase with a consequent reactivity decrease. The likelihood of local excesses of either dissolved or dispersed UF_3 does not appear to be large.

If, as seems very unlikely, the leak persisted, undetected, until 20 to 25 lb of NaK had entered the fuel stream, essentially all the uranium would be present as UF_3 (one fourth dissolved, the rest suspended). At this stage, metallic zirconium would begin to precipitate according to reaction (2) above. No metallic uranium could be formed until this reaction was nearly completed; almost 1000 lb of NaK would be required.

The case of a slow leakage of NaK into a NaF-KF-LiF-UF_4 fuel mixture would be somewhat different. The first product would again be UF_3, which appears to be quite soluble in this solvent. However, by the time about 15 lb of NaK had been added, the disproportionation reaction

$$4UF_3 \rightleftharpoons 3UF_4 + U^o$$

would begin to deposit metallic uranium in the hottest part of the system. The reaction would reverse as the material cooled in the heat exchanger. The extent to which deposition of the U^o from the very rapidly flowing stream and alloying of this metal with the walls would proceed is not known; losses of U^o from this source would, presumably, occur at the reactor exit headers and the heat exchanger inlet. It appears, therefore, that, in addition to the dilution by NaK, reactivity losses due to loss of U^o from the stream would occur.

Large leaks from the heat exchanger are somewhat more difficult to evaluate. If the fuel system were NaF-KF-LiF-UF_4, local excesses of NaK would precipitate U^o from the solution at or near the leak source. The heat exchanger would hardly be expected to plug, since alloying of Inconel with the U^o might weaken the tubing and enlarge the leak. No molten U^o could, apparently, be formed by the heat of reaction. While the net effect of the leak would be fuel dilution, and while no mechanism by which the core could pick up uranium metal is apparent, it is difficult to specify that no local high concentrations (as suspended U^o) could be sent through the core. Some heat of reaction would, of course, be realized, so the fuel entering the core should be hotter than normal; while this is in the favorable direction, the magnitude of the effect is very difficult to state for the general case.

If the fuel were NaF-ZrF_4-UF_4, then Zr^o and U^o would be co-precipitated and the heat of reaction might produce liquid U^o or, more likely, a U-Zr alloy. It is possible, but by no means certain, that plugging of a portion of the heat exchanger with NaF, U^o, and Zr^o could occur. Again the leak would result in fuel dilution, but it is possible that local excesses of

uranium could enter the core. In addition, the high heat of reaction might cause further disruption of the heat exchanger and increased leakage. The complex picture is further complicated by possible boiling of ZrF_4 from fuel in the vicinity.

Leakage of fuel into the heat exchanger tubes would result in fission-product activity in the NaK stream outside the reactor container, and therefore it would be immediately observed and more immediately troublesome but, in general, less likely to result in nuclear accidents.

Leakage of fuel into the NaK stream would result in immediate production of U^o, with either fuel, as well as Zr^o, if ZrF_4 were present, and the small coolant tubes would be likely to plug if the fuel were NaF-ZrF_4-UF_4. A large leak of NaF-ZrF_4-UF_4 would yield high melting materials which might plug the air radiator. It appears likely that most of the precipitated metals would be carried to the radiator and that very high fission-product activity would prevail at the radiator.

Reactions from a Core Leak. If leakage of the Na^o from the moderator coolant into the fuel system occurred, the results would be quite similar, in all cases, to those described above for leakage from the heat exchanger.

Leakage of fuel into the moderator coolant stream, however, would be more obviously dangerous, since by this type of failure more than the normal quantity of uranium would certainly be admitted to the core. In this case plugging of the leak could hardly be expected, since fission heating of the precipitated uranium would cause melting of this metal (and perhaps local melting of Be^o as well).

Conclusions

In order to minimize the corrosion from the fluoride fuels the fluoride mixture will be carefully pretreated to reduce the NiF_2 and FeF_2 impurities which are responsible for most of the 4 mils of initial corrosion. In addition, the mixture will contain significant amounts of the uranium in the form of UF_3 rather than UF_4 to minimize the mass transfer of the chromium. Under these conditions the corrosion of Inconel by the NaF-ZrF_4-UF_4 fuel in thermal-convection loops is about 6 mils in 1000 hr and should be in the neighborhood of 10 to 15 mils in the ART unless the higher flow velocities have a much greater effect on mass transfer than can be hypothesized. Corrosion by the NaF-KF-LiF-UF_4 system, which is not as well understood as that by the NaF-ZrF_4-UF_4 fuel, is now being carefully examined.

Only very rapid mixing of near stoichiometric quantities of Na^o or NaK with ZrF_4-bearing fuels under adiabatic conditions, which are very unlikely, can produce molten uranium metal.

While any detectable leak would, presumably, terminate the experiment, it appears that slow leaks (of the order of 100 g/min) would not be immediately dangerous and could be promptly detected. Rapid leakage into or

out of either the NaK or the fuel circuits would be more troublesome to evaluate and potentially more immediately dangerous.

Leakage of the moderator-coolant circuit will, insofar as possible, be in the direction such that Na^{O} would enter the fuel circuit. While it might be possible, if the leakage were considerable, for local excesses of uranium to enter the core region before corrective measures could be taken, leakage in the reverse direction would make this more likely.

The preferred direction of leakage from the heat exchanger section is somewhat more difficult to assess. A leak of fuel into the NaK stream would certainly result in high fission-product activity at the air radiator (that is outside the reactor assembly container) but would not be likely to result in a nuclear "incident." A large leak of NaK into the fuel might, but is not certain to, permit more than the tolerable concentration of uranium to enter the core. A proper balancing of these conflicting demands must be based on the inherent ability of the system to avoid nuclear accidents and the ability of the reactor cell to contain them if they occur. At the present time it is believed that it will be best to contain the activity within the reactor cell, and therefore the NaK pressure in the heat exchanger is to be kept above that of the fuel.

Appendix D

EXTREME NUCLEAR ACCIDENT - ANALYTICAL SOLUTION

A computation of the power excursion during rapid introduction of fuel into the reactor was made. The calculation was performed first for the general case, but a specific example was carried out concurrently. The numbers in brackets at the right margin below an equation refer to this example. The example is believed to represent a very severe set of assumptions, and its purpose is to convey a feeling for the order of magnitude of the quantities involved. At the end, a discussion of the effects of certain variations in parameters on the power excursion is presented.

General Equations

Assume a reactor of effective length H_r (30 in.) and a cross section A_r (173 in.^{2}). The reactor power is

$$P = y P_o \quad . \quad . \quad . \quad ,$$

where P_o is the stationary power in watts (6×10^7). The power density is

$$\frac{y P_o \text{ (w)}}{H_r \text{ (in.)} \times A_r \text{ (in.}^2\text{)} \times 16.4 \text{ (cm}^3\text{/in.}^3\text{)}} = \frac{1}{16.4} \frac{y P_o}{H_r A_r} \text{ (w/cm}^3\text{)} \quad ,$$

$$\left[705\ y\ (\text{w/cm}^3)\right] \quad .$$

The fuel has a specific heat, C_p, of 0.23 cal/g·°C, and a density, ρ, of 3.16 g/cm^3. Hence the temperature rise is

$$\frac{1}{16.4} \frac{y P_o}{H_r A_r} \text{ (w/cm}^3\text{)} \frac{1}{C_p \text{ (cal/g·°C)} \times 4.185 \text{ (w·sec/cal)} \times \rho \text{ (g/cm}^3\text{)}}$$

$$= \frac{1}{68.63} \frac{y P_o}{H_r A_r C_p \rho} \text{ (°C/sec)} \quad ,$$

$$\left[232\ y\ (^\circ\text{C/sec})\right] \quad .$$

With a fuel volume expansion coefficient, β, of 2.5×10^{-4} per ^{o}C the fuel volume expelled from the reactor per second is

$$\frac{1}{68.63} \frac{y\, P_o\ (^{o}C/sec)\ \beta \times H_r A_r\ (in.^3)}{H_r A_r\, C_p\, \rho\ (^{o}C) \times 1728\ (in.^3/ft^3)} = \frac{10^{-5}}{1.18} \frac{\beta\, P_o}{C_p\, \rho}\ y\ (ft^3/sec) \ ,$$

$$\left[0.175\ y\ (ft^3/sec)\right] .$$

With a rate of fuel flow into the reactor of $F = 2.7\ ft^3/sec$, the net flow into the reactor is

$$\left(F - \frac{10^{-5}}{1.18} \frac{\beta\, P_o}{C_p\, \rho}\, y\right)\ (ft^3/sec) \ ,$$

$$\left[2.7 - 0.175\ y\ (ft^3/sec)\right] .$$

If this net flow is divided by the volume of the reactor to get the fractional change $\Delta M/M$ of the fuel in the reactor, and $k_{ex}/(\Delta M/M) = 0.25$ is assumed:

$$\frac{d\, k_{ex}}{dt} = \frac{0.25 \times 1728\ (in.^3/ft^3)}{H_r A_r\ (in.^3)} \left(F - \frac{10^{-5}}{1.18} \frac{\beta\, P_o}{C_p\, \rho}\, y\right)\ (ft^3/sec)$$

$$= \left(\frac{432}{H_r A_r} F - 3.66 \times 10^{-3} \frac{\beta\, P_o}{H_r A_r\, C_p\, \rho}\, y\right)\ (sec^{-1}) \ ,$$

$$\left[0.225 - 0.0146\ y\ (sec^{-1})\right] .$$

The reactor power $y\, P_o$ is given by

$$\frac{d \ln (y\, P_o)}{dt} = \frac{k_{ex}}{\tau} \ ,$$

where τ is the prompt neutron life time (10^{-4} sec), and

$$\frac{d^2 \ln y}{dt^2} = \frac{432\, F}{H_r A_r\, \tau} - 3.66 \times 10^{-3} \frac{\beta\, P_o}{H_r A_r\, C_p\, \rho\, \tau}\ y = a - by\ (sec^{-2}) \ ,$$

$$\left[2250 - 146\ y\ (sec^{-2})\right] .$$

If ln y = z is substituted, the equation becomes

$$\frac{d^2z}{dt^2} + be^z - a = 0 \quad .$$

Multiply by dz/dt and integrate over t:

$$\frac{1}{2}\left(\frac{dz}{dt}\right)^2 + b(e^z - 1) - az = 0 \quad ,$$

where the constant of integration has been chosen in such a way that at t = 0, y = 1 (z = 0) and k_{ex} (and hence dz/dt) = 0. Hence,

$$\frac{dz}{dt} = \sqrt{2\left[az - b(e^z - 1)\right]} \quad ,$$

and

$$\int_0^z \frac{d\eta}{\sqrt{2\left[a\eta - b(e^\eta - 1)\right]}} = t \quad .$$

This equation can be solved by graphical integration, except for the part near z = 0. In this region, the integrand becomes infinite; however, the following expansion is an adequate approximation:

$$t \sim \int_0^z \frac{d\eta}{\sqrt{2(a-b)\eta}} = \sqrt{\frac{2z}{(a-b)}}$$

$$\left[t = 0.0308\sqrt{z}\right] .$$

The term $\dot{z}$, and, hence, the reactor power, assumes a maximum at

$$0 = \left.\frac{dz}{dt}\right|_{z=z_m} = \sqrt{2\left[az_m - b(e^{z_m} - 1)\right]} \quad ,$$

$$z_m = \frac{b}{a}(e^{z_m} - 1) \quad ,$$

$$= 0.06488\,(e^{z_m} - 1) \quad ,$$

$$\left[z_m = 4.181\right] .$$

The graphical integration again breaks down, because the integrand becomes infinite. Here the expansion

$$a\eta - b(e^{\eta} - 1) = (\eta - z_m)\left.\frac{dz}{d\eta}\right|_{\eta=z_m} = (\eta - z_m)(a - b\,e^{z_m}) = 7302\,(4.181 - \eta)$$

can be used. Near the maximum

$$t_m - t = \int_z^{z_m} \frac{d\eta}{\sqrt{2\,[a\eta - b(e^{\eta} - 1)]}} \approx \frac{1}{\sqrt{2\,(b\,e^{z_m} - a)}} \int_z^{z_m} \frac{d\eta}{\sqrt{z_m - \eta}}$$

$$= \frac{1}{\sqrt{2(be^{z_m} - a)}} \left[2\sqrt{z_m - z}\right],$$

$$\left[t_m - t = \sqrt{\frac{2}{7302.98}}\sqrt{z_m - z}\right].$$

In Fig. D.1 is shown the resultant curve for the example given.

The pressure p can be computed under the assumption that the expanding fuel has to accelerate a column 30 in. long having a cross sectional area equal to that of the inlet and outlet ducts. The point of pressure relief may not be at the end of the core inlet or outlet ducts, and for this reason it would appear that the column might be longer. On the other hand, some of the fuel will be quite close to the reactor outlet and will have to accelerate only a short column. Hence the above assumption will not be too far off for the average.

A temperature rise (dT/dt) causes fuel to be expelled with a velocity $v = \beta H_r (dT/dt)$. The derivative of this, multiplied by the mass of 1 cm^2 of the column to be accelerated, gives the pressure:

$$p = \beta H_r^2 \rho \frac{d^2T}{dt^2} \times 6.45 \times 10^{-6}$$

$$= 6.45 \times 10^{-6}\, \beta H_r^2 \rho \frac{1}{68.63\, H_r A_r C_p \rho} \frac{dy\, P_o}{dt}$$

$$= 0.9 \times 10^{-7} \frac{\beta H_r}{A_r C_p} \frac{P}{\mathscr{H}},$$

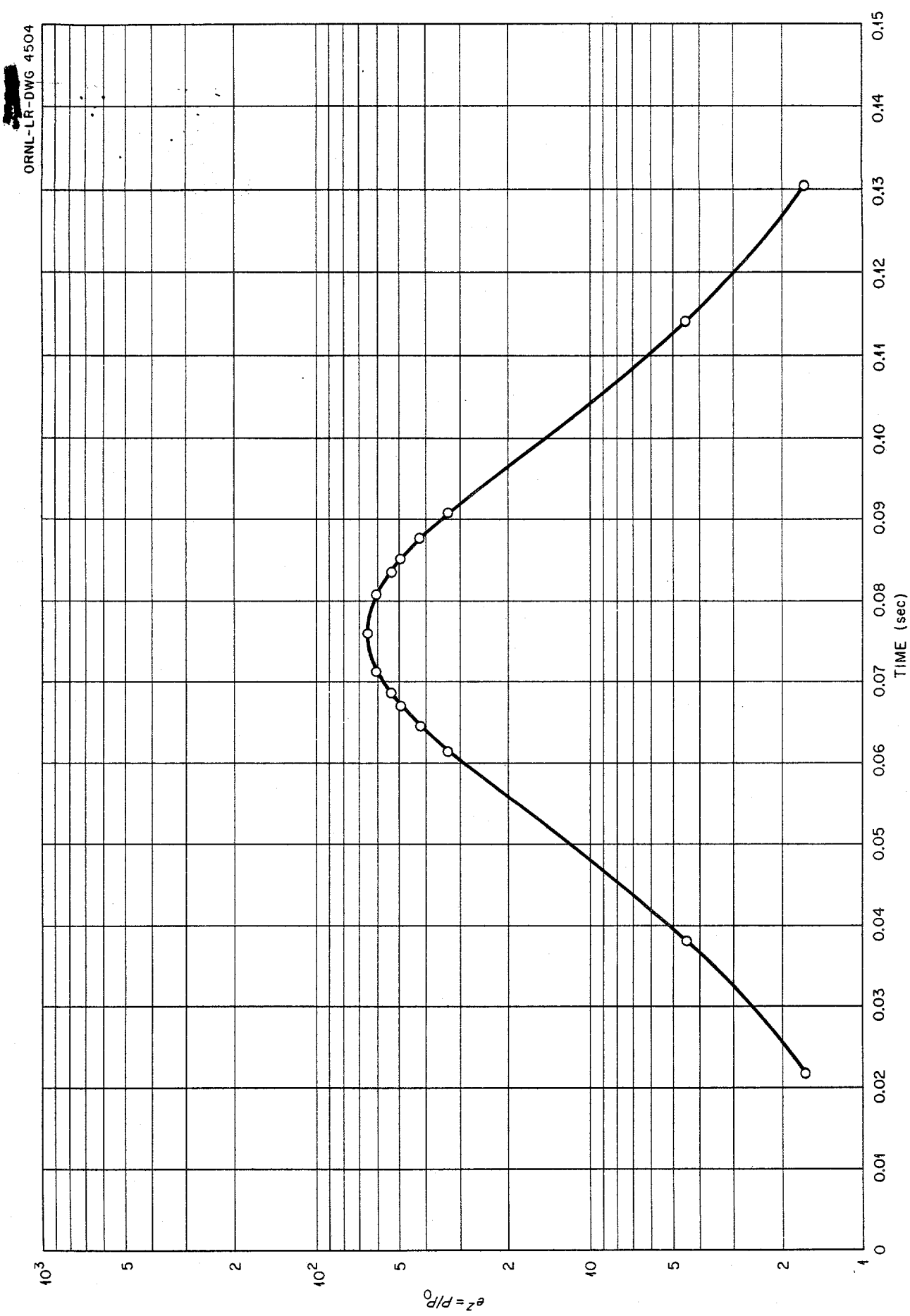

Fig. D-4. Reactor Power P Relative to the Initial Power P_0 vs. Time During Excursion.

where $\mathcal{H}$ is the reactor period. From Fig. D.1, it can be seen that $P/\mathcal{H}$ can reach 2×10^{11} w/sec. Hence, the pressure in the example would be

$$0.9 \times 10^{-7} \times \frac{2.5 \times 10^{-4} \times 30}{173 \times 0.23} \times 2 \times 10^{11} = 3.5 \text{ atmospheres},$$

a very moderate value.

By graphical integration (see Fig. D.1) the maximum is reached 76.5 msec after the start of the disturbance. Up to that time, 60 x 1.191 Mw-sec have been put into the reactor. Since the reactor temperature goes up by 232°C/sec, at normal reactor power, the temperature rise up to the peak of the power curve is 232 x 1.191 = 276°C = 500°F; that is, the fuel is halfway to the boiling point.

After the maximum is reached, the power decreases according to a curve which is symmetrical to the ascent. The temperature continues to increase so that the boiling point is reached just about at the minimum of the power curve, where the power is P_o. Boiling at this power does not appear to be serious. It should be pointed out, however, that the above conclusion is a consequence of the parameters used. The consequences of variations in the parameters are discussed in the following.

Variation of Parameters. The shortest positive reactor period is reached when there is the greatest amount of fuel in the reactor. This occurs at the time when the fuel expulsion due to thermal expansion is equal to the flow of fuel into the reactor. Before that time the influx of fuel exceeds the efflux; after that time, the opposite is true. Hence the shortest period is reached when

$$F = \frac{10^{-5}}{1.18} \frac{\beta P_o}{C_p \rho} y \quad ,$$

$$y P_o = 1.18 \times 10^5 \frac{C_p \rho}{\beta} F \quad , \qquad [926 \text{ Mw}]$$

$$y = 1.18 \times 10^5 \frac{C_p \rho}{\beta} \frac{F}{P_o} = e^z = \frac{a}{b} \quad , \qquad [15.4]$$

The period is defined as the reciprocal of

$$\frac{d}{dt} (\ln y P_o) = \frac{dz}{dt} = \sqrt{2b} \sqrt{e^z(z-1) + 1}$$

$$= \sqrt{2 \times 3.66 \times 10^{-3} \frac{\beta P_o}{H_r A_r C_p \rho \tau}} \sqrt{e^z(z-1) + 1} \quad , \qquad [90] .$$

Hence in the example, the period is 1/90 sec = 11 msec.

It should be noted that the power at which the shortest period is reached does not depend on the initial power. It depends only on the material constant $C_p \rho/\beta$, and is proportional to the flow rate F. If $e^z(z-1) >> 1$, as will be the case in all instances in which there is any question of danger, the period becomes

$$\frac{1}{\sqrt{2a \left[\ln \frac{a}{b} - 1\right]}} .$$

Since a is independent of P_o, P_o enters into the period only through ln b, which is a weak dependence, so that the period is not very much shorter in the case of a startup accident. In the example, for P_o = 1% of the rated power, the period turns out to be 6 msec. The temperature attained at the time of the shortest period is almost the same, whether the initial power is the rated power or a fraction thereof. The temperature rise during the time until rated power is reached is very small.

At the power maximum,

$$\frac{dz}{dt} = 0 ,$$

and

$$\frac{a}{b} z_m = e^{z_m} - 1 .$$

Fig. D.2 shows e^{z_m} as a function of the parameter $\frac{a}{b} = 1.18 \times 10^5 \frac{C_p}{\beta} \frac{F}{P_o} \rho$.

For the values assumed in the previous discussion (a/b = 15.4), P_{max}/P_o is 65.5. However, for a startup accident, at which the original power was 1% of the rated power, a/b would be 1540, and P_{max}/P_o would be 14,500. This corresponds to a maximum power of 14,500 x 0.01 x 60 Mw = 8700 Mw. The reactor temperature reached at this point is not much different from the temperature reached at the power maximum in the previous example, because the initial phase of low power does not contribute much to the heating. The boiling point will not be reached until the power curve is on the downgrade.

Some of the constants in the calculations are, as yet, somewhat subject to doubt. For instance, τ, the mean life of the prompt neutrons cannot be readily measured. If this quantity had 4 times the assumed value, the time scale would be slowed down by a factor of 2 - other things remaining the same. Since a given power would then be reached in twice the time required in the previous example, the temperature rise would be twice as high. In particular, the maximum power would be reached just at the boiling point. A simultaneous reduction of the flow rate F and of β, or a reduction of $k_{ex}/(\Delta M/M)$ would have a similar effect.

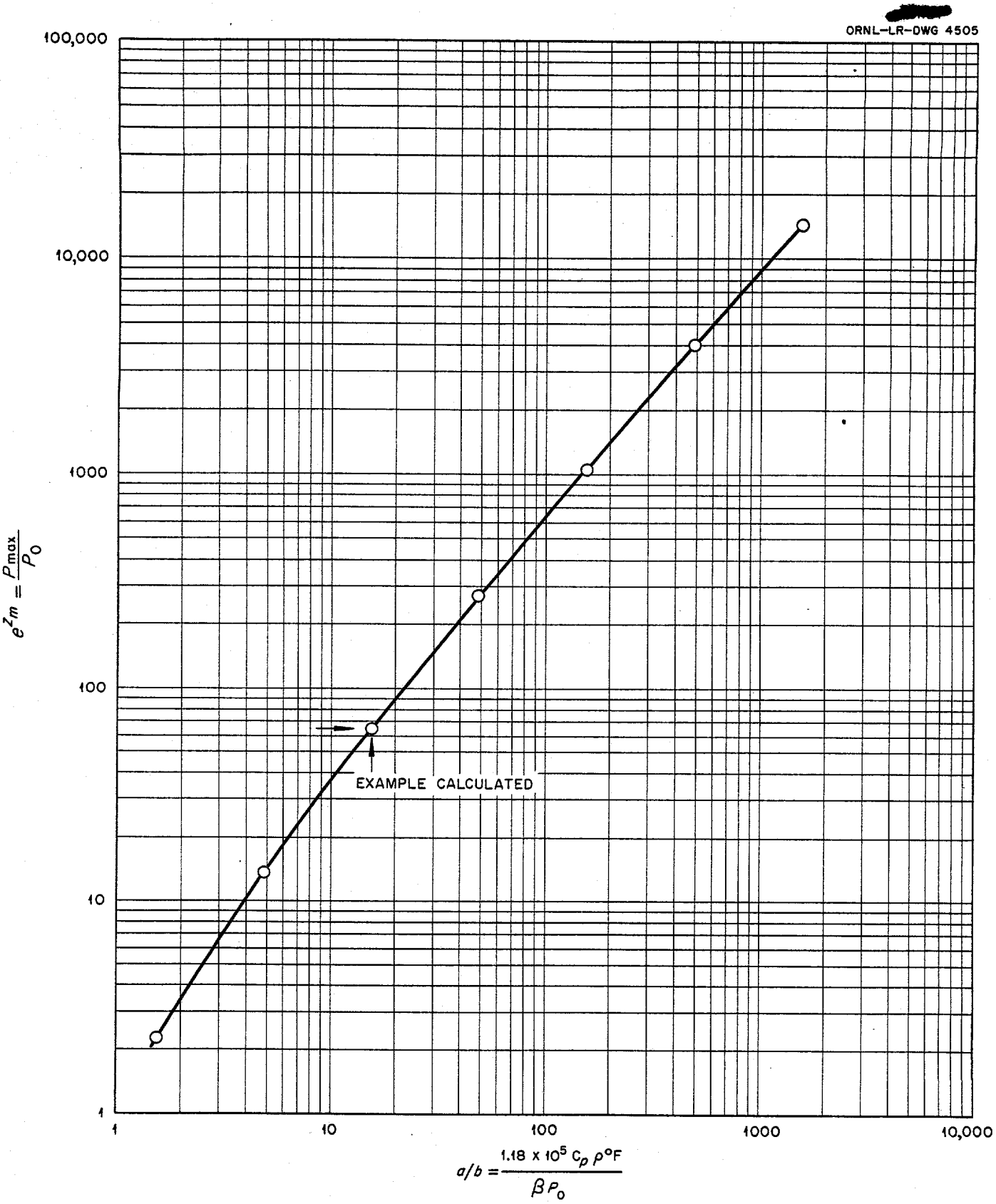

Fig. D-2. Effect of Parameters on Maximum Reactor Power.

From the above it follows that a combination of unfavorable values of the, as yet, incorrectly known constants and of a startup accident could lead to boiling at about 10,000 Mw. The consequences of boiling are discussed in Appendixes E and F.

Appendix E

EXTREME NUCLEAR ACCIDENT
NUMERICAL SOLUTION

In addition to the analytical solution to the problem of the extreme nuclear accident given in Appendix D, an independent numerical approach has also been employed. Agreement between the two methods is quite good. The numerical method of solution has been applied to a number of typical cases, and the data are tabulated in Tables E.2 through E.6. In most instances a 10-msec time interval was used. An expression for the change in reactivity for that time interval was then chosen for the type of excursion in question. In most instances this was taken as a constant (peculiar to the case in question) minus the loss in reactivity associated with thermal expansion, i.e.,

$$\Delta k_n' = a - \frac{\Delta V'_{n-1}}{5V} = a - \frac{\Delta V'_{n-1}}{15} \quad ,$$

where $\Delta V'$ is the thermal expansion of the fuel in the core during the time interval in question. Note that this expression presumes a core volume of 3 ft^3, and $(\Delta k/k) = (1/5)(\Delta M/M)$. The time t from the beginning of the excursion and the corresponding value for k_{ex} were tabulated first. The period T was then calculated by ignoring the delayed neutrons and assuming a neutron lifetime of 4×10^{-4} sec, i.e.,

$$T = \frac{4 \times 10^{-4}}{k_{ex}} \text{ (sec)} \quad .$$

The values given for T in Tables E.2 through E.6 have been multiplied by 1000 to express them in milliseconds. The power was computed from the period, i.e.,

$$P_n = P_{n-1}\, e^{\Delta t/T} \text{ (Mw)} \quad ,$$

where Δt was the time interval in msec, and P_n the power at the end of the time interval in question. Since 1 Mw equals approximately 1 Btu/msec, the net heat added to the fuel in the core during the time interval was taken as:

$$\Delta Q' = \left(\frac{P_n + P_{n-1}}{2} - 60 \right) \text{ Mw x 10 msec} \quad ,$$

where $\Delta Q'$ is in Btu and P is in Mw. The temperature increase ΔT during the time interval may then be obtained from the core volume, density, and specific heat of the fuel, i.e.,

$$\Delta T' = \frac{\Delta Q'}{W C_p V} = \frac{\Delta Q' \text{ (Btu)}}{200 \text{ (lb/ft}^3\text{)} \times 0.3 \text{ (Btu/lb}\cdot{}^\circ\text{F)} \times 3 \text{ (ft}^3\text{)}} = \frac{\Delta Q'}{180} \text{ (}^\circ\text{F)} .$$

The temperature increase over the normal value then becomes the summation of $\Delta T'$,

$$\Delta T_n = \sum_{t=o}^{t=n} \Delta T' \text{ (}^\circ\text{F)} ,$$

where n is the n^{th} time interval. Since an initial mean fuel temperature in the core of 1400°F was assumed, the mean fuel temperature follows directly. The volume increase $\Delta V'$ (ft^3) is simply,

$$\Delta V' = \beta V \Delta T' = 5.7 \times 10^{-4} \Delta T' \text{ (ft}^3\text{)} .$$

The total volume ΔV expelled from the core by thermal expansion is, then,

$$\Delta V_n = \sum_{t=o}^{t=n} \Delta V' \text{ (ft}^3\text{)} .$$

Since the fuel is essentially incompressible relative to the comparatively elastic pressure shell, expansion of the fuel in the core means either a rapid increase in pressure or, if a frangible diaphragm is ruptured, a buildup of the heat required to eject the fuel through the resulting orifice. As a representative case, a 2.16-in.-dia orifice was assumed to represent the passages through the fuel pumps to the expansion tank. The velocity through the orifice (in fps) is then

$$\text{Velocity, } u = \frac{\Delta V'}{A \Delta t} = 4000 \, \Delta V' \text{ (fps)}, \qquad \text{for } \Delta t = 10 \text{ msec.}$$

The pressure drop across the orifice required to eject the fluid is then

$$\Delta P_1 = \frac{W u^2}{2g \times 144} = \frac{200 \text{ (lb/ft}^3\text{)} \left[u \text{ (fps)}\right]^2}{64.4 \left(\frac{\text{lb mass}}{\text{lb fuel}}\right) \times 144 \text{ (in.}^2\text{/ft}^2\text{)}} = 0.0216 \left(\frac{\Delta V'}{A \Delta t}\right)^2 \text{(psi)} .$$

The pressure drop for any other orifice size will then be inversely proportional to the fourth power of the orifice diameter.

If the fuel is ejected into a channel having the same diameter as the orifice, it has to be accelerated by the pressure ahead of the orifice. The pressure required for this acceleration can be calculated from the change in momentum of the column, i.e.,

$$\Delta P_2 = \frac{1}{144}\left(\frac{\Delta V_{n-1} W}{g}\right)\left(\frac{\Delta V'_n}{A} - \frac{\Delta V'_{n-1}}{A}\right)\frac{1}{\Delta t} = 0.0431\,\frac{\Delta V_{n-1}}{\Delta t}\left[\frac{\Delta V'_n}{A\Delta t} - \frac{\Delta V'_{n-1}}{A\Delta t}\right].$$

The results of a series of numerical analyses are presented in Fig. E.1, and the tabulated data are given in Tables E.2 through E.5. In each instance it was assumed that the reactivity was increased at an initially constant rate and that a deduction from this rate should be made as the excursion progressed to allow for the rate at which reactivity would be lost from the core through fuel expulsion by thermal expansion. Curves were obtained for the very high initial rates of increase in reactivity; namely, 5, 10, 20, and 40% per second. An initial power level of 60 Mw was assumed, together with the fuel and reactor properties given in Table E.1.

Table E.1

Fuel and Reactor Properties at 60 Mw

Reactor core volume, ft^3	3	3
Mean neutron lifetime, sec	4×10^{-4}	4×10^{-4}
Fuel composition, mole %	50-NaF, 46-ZrF_4, 4-UF_4	11-NaF, 42-KF, 44-LiF, 3-UF_4
Fuel density, lb/ft^3	200	132
Fuel specific heat, Btu/lb·°F	0.27	0.41
Btu/ft^3·°F	55	55
Fuel volumetric coefficient of expansion, ft^3/ft^3·°F	1.9×10^{-4}	2.5×10^{-4}

The curves of Fig. E.2 show the effects of abrupt introduction of fuel into the moderator cooling passages, as discussed in Chapter 5, "Reactor Hazards." The calculations for this case are presented in Table E.6. Relief of the nuclear excursion through boiling of the fuel in the moderator-cooling passages is presumed. Note that the fuel in the core does not reach the boiling point.

Curves for the vapor pressure of the two typical fuels are given in Fig. E.3. In examining Table E.1 in light of the derivations developed in Appendix D, it is evident that for the zirconium and lithium fuels the parameter $(C_p\rho/\beta)$ is inversely proportional to their coefficients of thermal expansion. For this reason the calculations were carried out for the zirconium fuel with the thought that the resulting curves would correspond to similar curves for a lithium fuel with a 30% higher rate of increase in reactivity.

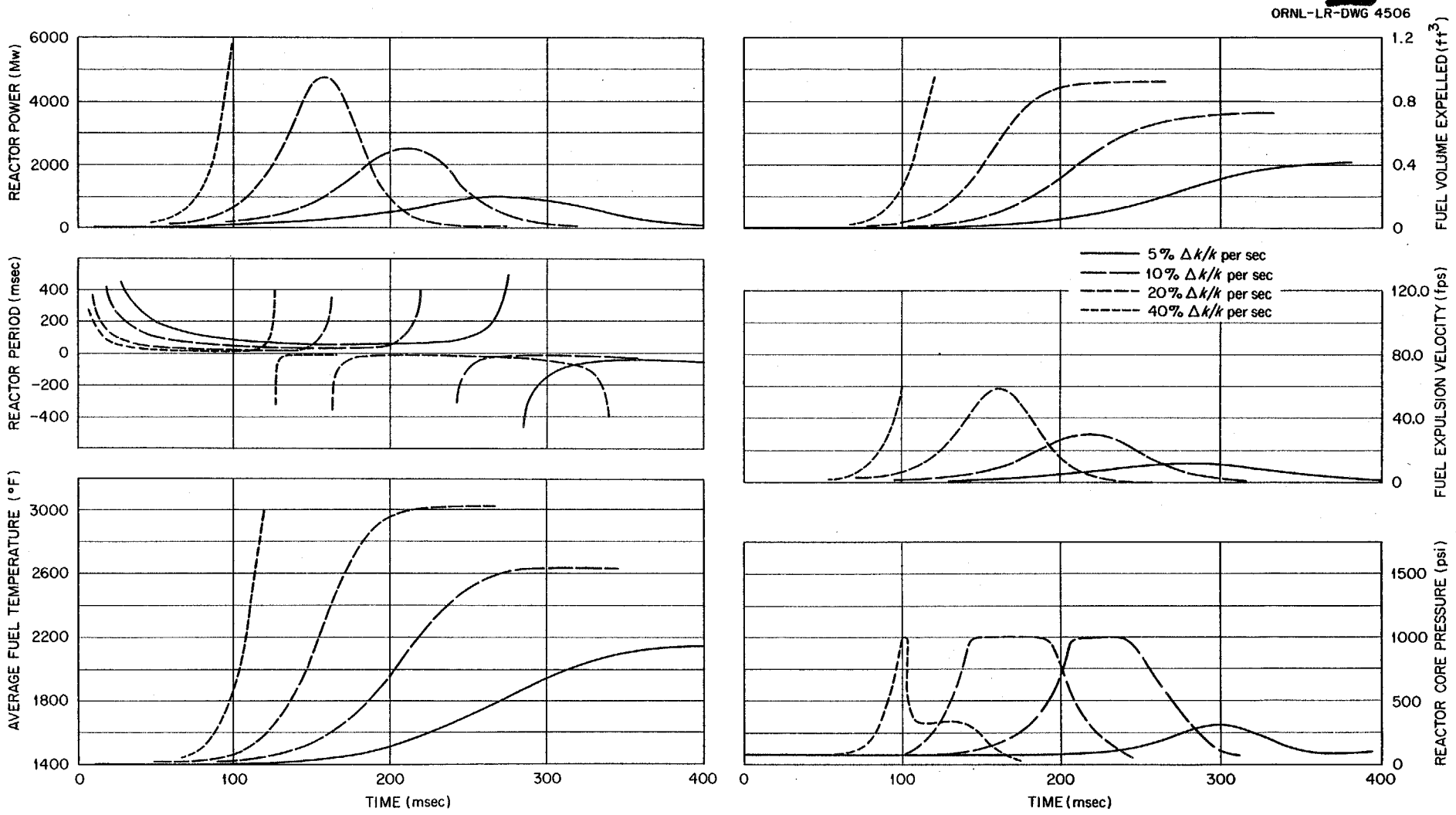

Fig. E-1. Nuclear Excursions for Fuel Deposition in the Core Increasing k_{eff} at Rates from 5% to 40% $\Delta k/k$ per sec.

ORNL-LR-DWG 4507

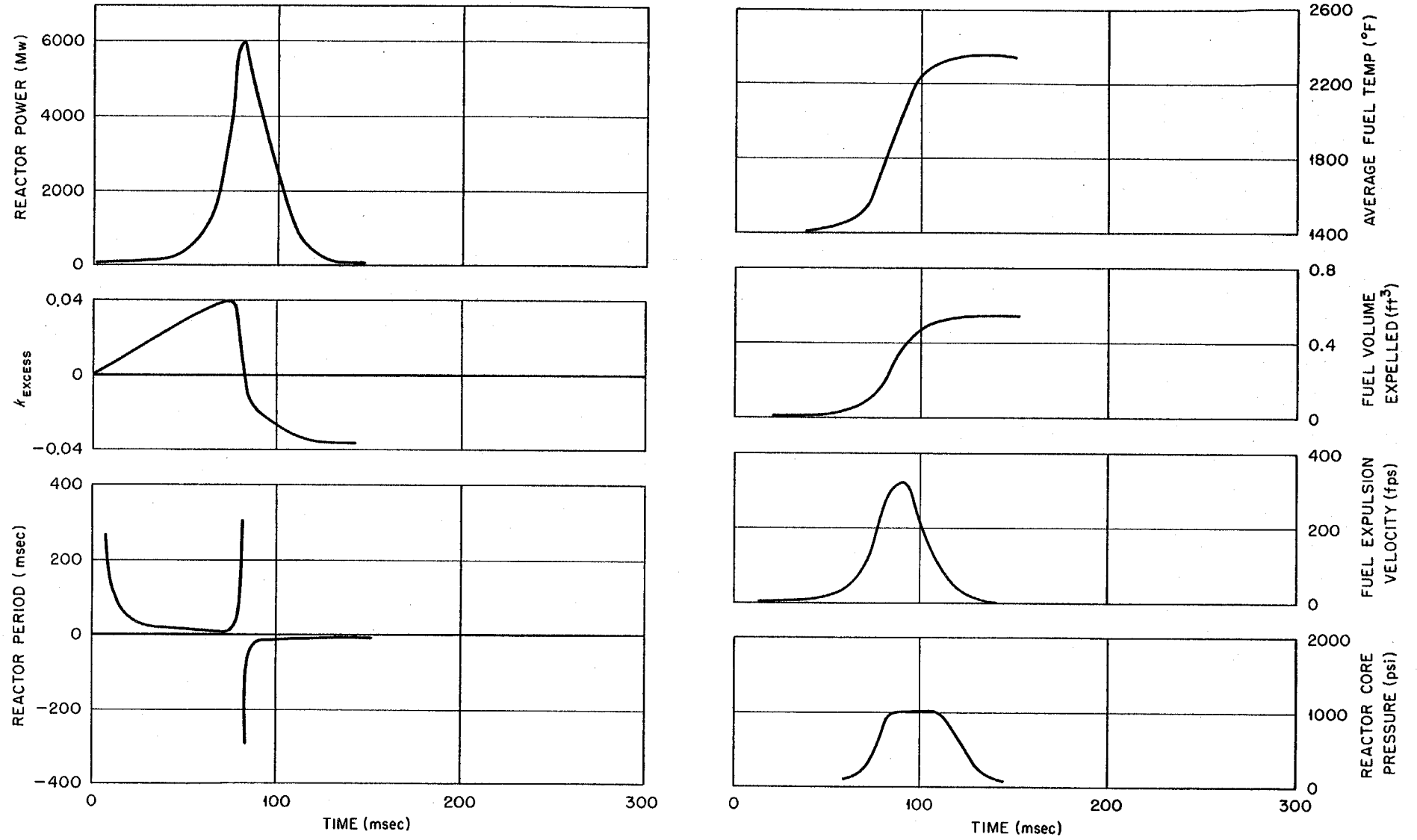

Fig. E-2. Nuclear Excursions for Fuel Entering the Moderator Cooling Passages and Increasing k_{eff} at the Rate of 60% $\Delta k/k$ per sec.

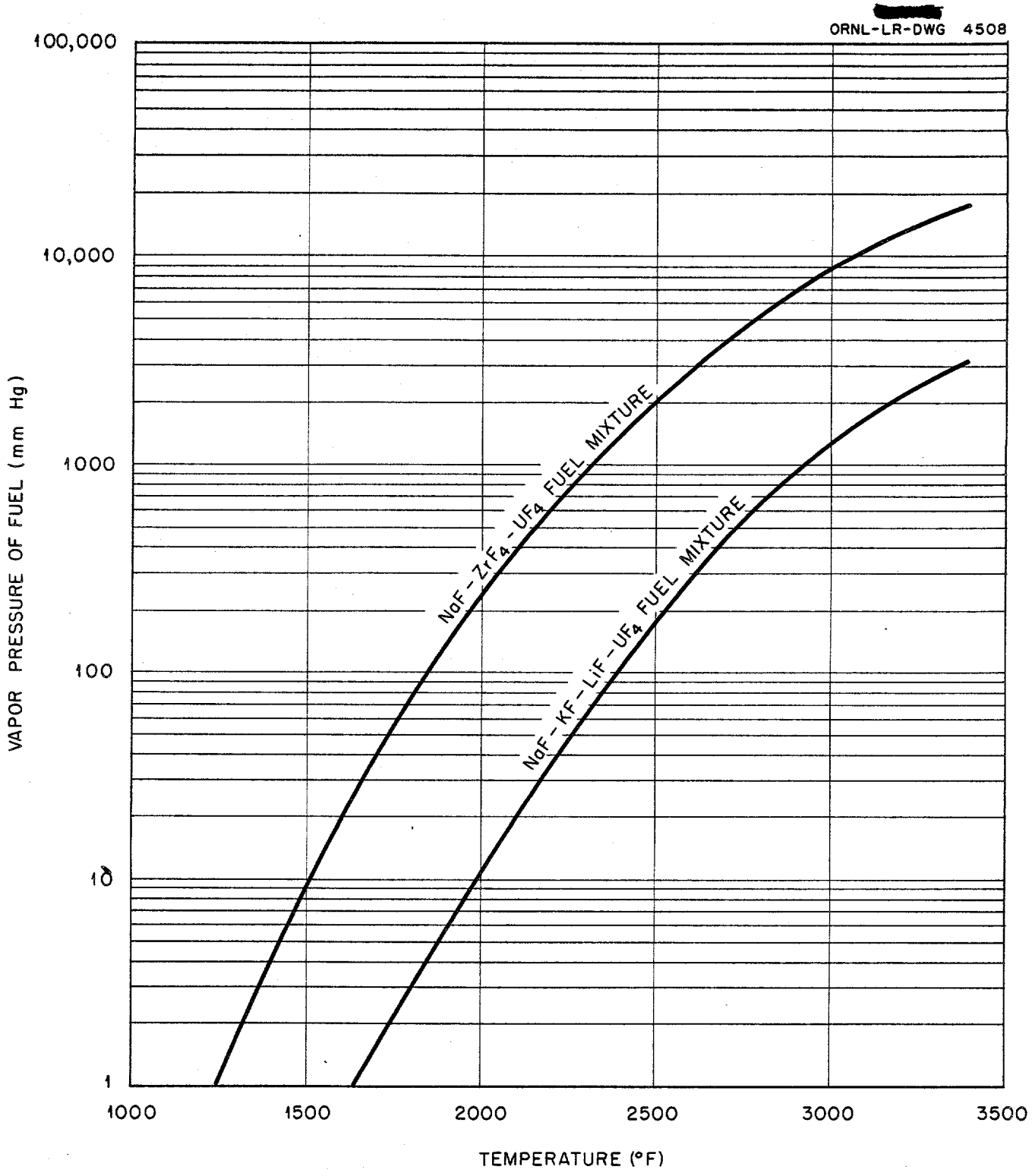

Fig. E-3. Fuel Vapor Pressure as a Function of Temperature.

The increase in fuel volume was considered as being absorbed in three different ways. Passages between the main fuel system and the expansion tank equivalent in area to a 2.1-in.-dia flow nozzle are the primary relief mechanism. If the fuel pressure becomes large, dilation of the main pressure shell becomes appreciable, with the volume increase accomodated in this way being roughly 0.07 ft^3 at a pressure of 1000 psi (see Appendix F). At 1000 psi, yielding of the pressure shell in a deliberately weakened zone just inside the main heat exchanger NaK outlet belt will begin and will provide for enough increase in volume to hold the pressure to about 1000 psi. It was considered that this yielding in shear would continue until the disk had moved outward about 0.25 in. to give an increase in volume of about 0.1 ft^3. Further expansion would cause rupture, and the Inconel disk sheared out of the pressure shell would then push through the thermal insulation and begin to push a lead disk out through the bottom of the gamma shield. These types of accident are illustrated in the calculations of Tables E.2 through E.6, with the severity of the accident depending on the initial rate of increase of k_{eff}. The pressure under the most extreme condition is relieved very rapidly by blowout of the bottom of the pressure shell, accompanied by boiling of a portion of the fuel in the core. A case of this sort is covered in Table E.5. Note that an additional column had to be added to calculate the force required to accelerate the Inconel and lead disks.

In examining the curves of Fig. E.1 it appears that, for the conditions assumed, an initial rate of increase in reactivity of 5%/sec could be tolerated without generation of excessive temperatures and pressures. Since this case would correspond to that in which 25% of the uranium would be separated from the fuel stream passing through the core and retained there while fresh fuel continued to enter at the normal rate, it appears to represent an extremely severe case.

TABLE E-2. NUCLEAR EXCURSION CALCULATION FOR A HYPOTHETICAL CAS

Conditions: Accident relieve
Fuel: $NaF\text{-}ZrF_4\text{-}UF_4$
Time Interval: 0.020 sec
Mean Neutron Lifetime: 4 ×

(a) Time, t (msec): $\Delta t = 20$	(b) k Excess, k_{ex}: $0.00005\,[(a) - (\Delta t/2)] - (j)_{n-1}/15$	(c) Period, τ (msec): $0.4/(b)$	(d) Power, P (Mw): $(d)_{n-1}\, e^{\Delta t/(c)}$	(e) Net Heat per Interval from Excursion, Q' (Btu): $[(d)_n + (d)_{n-1} - 120]\,\Delta t/2$	(f) Temperature Rise in Time Interval, $\Delta T'$ (°F): $(e)/180$	(g) Fuel Temperature Rise, ΔT_n (°F): Σ (f)	Mea T
0	0	∞	60	0	0	0	
20	0.0005	800	61.5	15	0.08	0.08	
40	0.0015	267	66.4	79	0.44	0.52	
60	0.0025	160	75.4	218	1.20	1.72	
80	0.0034	118	89.0	444	2.50	4.22	
100	0.0043	93	110.0	790	4.40	8.62	
120	0.0051	78	142	1,320	7.30	15.9	
140	0.0058	69	190	2,120	10.8	26.7	
160	0.0064	62.5	262	3,320	18.5	45.2	
180	0.0067	59.7	366	5,080	28.2	73.4	
200	0.0066	60.5	500	7,460	41.4	114.8	
220	0.0060	66.7	675	10,550	58.6	173.4	
240	0.0048	84	856	14,110	78.5	251.9	
260	0.0028	143	980	17,160	95.0	346.9	
280	0.0002	2000	990	18,500	103.0	449.9	
300	−0.0027	−148	860	17,300	96.0	545.9	
320	−0.00535	−75	658	13,980	88.0	633.9	
340	−0.00769	−52	448	9,860	55.0	688.9	
360	−0.0088	−45.5	299	6,270	35.0	723.9	
380	−0.0091	−44.0	189	3,680	20.5	744.4	
400	−0.0089	−44.8	121	1,900	10.5	754.9	
420	−0.0083	−48	80	810	5.0	760	

VOLVING FUEL DEPOSITION IN THE CORE TO GIVE AN INITIAL RATE OF INCREASE IN k_{eff} OF 5%/sec

fuel expulsion from core forced by thermal expansion

sec

(h)	(i)	(j)	(k)	(l)	(m)	(n)
el Temperature, °): (g) + 1400	Change in Fuel Volume in Time Interval, $\Delta V'$ (ft^3): (f) × 5.7 × 10^{-4}	Total Change in Fuel Volume, ΔV_n (ft^3): (g) × 5.7 × 10^{-4}	Change in Volume Absorbed in Pressure Shell Dilation, ΔV_D (ft^3): 7.6 × 10^{-5} [$(n)_n - (n)_{n-1}$]	Change in Volume Absorbed at Blowout Disk, ΔV_e (ft^3)	Fuel Exit Velocity, u (fps): [(i) − (k) − (l)] × 10^3/(Δt × 0.025)	Velocity Pressure in 2.1-in.-dia Jet, ΔP (psi): 0.0216 $(k)^2$
1400	0	0	0	0		
1400	0.000046	0.000046	0			
1401	0.00025	0.00030	0		0.5	
1402	0.00069	0.00098	0		1	
1404	0.00142	0.00240	0		3	
1408	0.00255	0.00492	0		5	
1416	0.00410	0.00910	0.0001		8	1.38
1427	0.00616	0.0153	0.0001		12	3.11
1445	0.0106	0.0260	0.0005		21	9.52
1473	0.0161	0.0420	0.0010		32	22.0
1515	0.0236	0.0652	0.0019		47	47.5
1513	0.0335	0.0990	0.0024		60	78
1652	0.0447	0.144	0.0046		80	140
1747	0.0540	0.198	0.0058		100	216
1850	0.0590	0.260	0.0036		110	263
1946	0.0548	0.312	0.0000		110	261
2034	0.0501	0.361	−0.0022		105	235
2089	0.0314	0.392	−0.0078		79	133
2124	0.0200	0.411	−0.0056		50	56
2144	0.0117	0.425	−0.0039		29	19
2155	0.00600				12	3.11
2160						

TABLE E-3. NUCLEAR EXCURSION CALCULATION FOR A HYPOTHETICAL CASE INVOLVING FUEL

Conditions: Accident relieved by fuel expulsion
Fuel: $NaF\text{-}ZrF_4\text{-}UF_4$
Time Interval: 0.020 sec
Mean Neutron Lifetime: 4×10^{-4} sec

(a) Time, t (msec): $\Delta t = 20$	(b) k Excess, k_{ex}: $0.0001\ [(a) - (\Delta t/2)] - (j)_{n-1}/15$	(c) Period, τ (msec): $0.4/(b)$	(d) Power, P (Mw): $(d)_{n-1}\, e^{\Delta t/(c)}$	(e) Net Heat per Interval from Excursion, Q' (Btu): $[(d)_n + (d)_{n-1} - 120]\ \Delta t/2$	(f) Temperature Rise in Time Interval, $\Delta T'$ (°F): $(e)/180$	(g) Fuel Temperature Rise, ΔT_n (°F): $\Sigma\ (f)$	(h) Mean Fuel Temperature, T_F (°F): $(g) + 1400$
0	0.060	∞	60	0	0	0	1400
20	0.001	400	63	30	0.17	0.17	1400
40	0.00299	134	73.1	161	0.89	1.06	1401
60	0.00496	80.6	93.7	468	2.60	3.66	1404
80	0.00686	58.3	132	1,057	5.87	9.53	1410
100	0.00864	46.3	203	2,150	11.9	21.4	1421
120	0.01019	39.3	338	4,210	23.4	44.8	1445
140	0.01130	35.4	595	8,130	45.2	90.0	1490
160	0.01158	34.5	1062	15,370	85.4	175	1575
180	0.01033	38.7	1781	27,230	151	326	1726
200	0.00659	60.7	2476	41,370	230	556	1956
220	−0.00014	−2857	2459	48,150	268	824	2224
240	−0.00834	−48	1621	39,600	220	1044	2444
260	−0.01467	−27.3	779	22,800	127	1171	2571
280	−0.01250	−22.9	325	9,840	54.7	1226	2626
300	−0.01758	−22.8	135	3,400	18.9	1245	2645
320	−0.01630	−24.5	59.7	747	4.15	1249	2649

SITION IN THE CORE TO GIVE AN INITIAL RATE OF INCREASE IN k_{eff} OF 10%/sec

re forced by thermal expansion

(i)	(j)	(k)	(l)	(m)	(n)
ange in Fuel lume in Time val, $\Delta V'$ (ft^3): $\times 5.7 \times 10^{-4}$	Total Change in Fuel Volume, ΔV_n (ft^3): $(g) \times 5.7 \times 10^{-4}$	Change in Volume Absorbed in Pressure Shell Dilation, ΔV_D (ft^3): $7.6 \times 10^{-5}\,[(n)_n - (n)_{n-1}]$	Change in Volume Absorbed at Blowout Disk, ΔV_e (ft^3)	Fuel Exit Velocity, u (fps): $[(i) - (k) - (l)] \times 10^3/(\Delta t \times 0.025)$	Velocity Pressure in 2.1-in.-dia Jet, ΔP (psi): $0.0216\ (k)^2$
0	0	0	0	0	0
0.000095	0.000093	0	0	0	0
0.00051	0.0006	0	0	1	0
0.00148	0.00209	0	0	3	0
0.00335	0.00543	0	0	7	1
0.00678	0.0122	0	0	14	4
0.0133	0.0255	0.0012	0	25	13
0.0258	0.0513	0.0022	0	45	44
0.0487	0.0998	0.0066	0	77	130
0.0861	0.186	0.025	0	124	330
0.131	0.317	0.030	0	196	650
0.153	0.470	0.011	0.033	215	1000
0.125	0.595	0.000	0.019	215	1000
0.0724	0.667	−0.021	0	185	740
0.0312	0.699	−0.032	0	127	325
0.0108	0.710	−0.017	0	58	75

TABLE E-4. NUCLEAR EXCURSION CALCULATION FOR A HYPOTHETICAL CAS

Conditions: Accident relie
Fuel: NaF-ZrF_4-UF_4
Time Interval: 0.010 sec
Mean Neutron Lifetime: 4 ×

(a) Time, t (msec): $\Delta t = 10$	(b) k Excess, k_{ex}: 0.0002 $[(a) - (\Delta t/2)]$ $-(j)_{n-1}/15$	(c) Period, τ (msec): 0.4/(b)	(d) Power, P (Mw): $(d)_{n-1}\, e^{\Delta t/(c)}$	(e) Net Heat per Interval from Excursion, Q' (Btu): $[(d)_n + (d)_{n-1} - 120]\, \Delta t/2$	(f) Temperature Rise in Time Interval, $\Delta T'$ (°F): (e)/180	(g) Fuel Temperature Rise ΔT_n (°F): Σ (f)
0	0	∞	60	0	0	0
10	0.001	400	61.5	7.5	0.041	0.041
20	0.003	133	65.2	33.5	0.185	0.226
30	0.005	80	73.8	95.0	0.53	0.756
40	0.007	57	87.7	208.0	1.16	1.92
50	0.0089	45	109.5	386	2.15	4.07
60	0.0108	37	143	665	3.70	7.77
70	0.0127	31.5	197	1,100	6.1	13.9
80	0.0144	27.8	283	1,800	10.0	23.9
90	0.0160	25.5	420	2,901	16.1	40.0
100	0.0173	23.1	648	4,740	26.4	66.4
110	0.0182	22.0	1020	7,740	43.0	109.4
120	0.0183	21.8	1610	12,650	70.0	180
130	0.0173	23.2	2480	19,850	110.0	290
140	0.0146	27.4	3570	29,700	165.0	455
150	0.0096	21.6	4530	39,900	222.0	677
160	0.0022	182	4780	45,950	256.0	933
170	−0.0066	−60.5	4070	43,650	242	1175
180	−0.0148	−27.1	2880	34,150	190	1365
190	−0.0208	−19.3	1720	22,400	124	1489
200	−0.0240	−16.7	941	12,700	71	1560
210	−0.0250	−16.0	502	6,600	37	1597
220	−0.0246	−16.3	272	3,270	18	1615
230	−0.0233	−17.2	152	1,520	8	1623
240	−0.0217	−18.5	89	655	4	1627
250	−0.0198	−20.2	54	116	0.6	1628
260	−0.0178	−22.5	35	−155	−0.9	1627

*Piece shears out of pressure shell through heat exchanger outlet bell.

VOLVING FUEL DEPOSITION IN THE CORE TO GIVE AN INITIAL RATE OF INCREASE IN k_{eff} OF 20%/sec

y fuel expulsion from core forced by thermal expansion

$^{-4}$ sec

(b)	(i)	(j)	(k)	(l)	(m)	(n)
ean Fuel Temperature, T_F (°F): (g) + 1400	Change in Fuel Volume in Time Interval, $\Delta V'$ (ft^3): $(f) \times 5.7 \times 10^{-4}$	Total Change in Fuel Volume, ΔV_n (ft^3): $(g) \times 5.7 \times 10^{-4}$	Change in Volume Absorbed in Pressure Shell Dilation, ΔV_D (ft^3): $7.6 \times 10^{-5}[(n)_n - (n)_{n-1}]$	Change in Volume Absorbed at Blowout Disk, ΔV_e (ft^3)	Fuel Exit Velocity, u (fps): $[(i) - (k) - (l)] \times 10^3/(\Delta t \times 0.025)$	Velocity Pressure in 2.1-in.-dia Jet, ΔP (psi): $0.0216\ (k)^2$
1400	0	0	0	0	0	0
1400	0.00002	0.00002	0	0	0.1	0
1400	0.00011	0.00013	0	0	0.4	0
1401	0.0003	0.0004	0	0	1.2	0.03
1402	0.00066	0.0011	0	0	2.64	0.15
1404	0.00123	0.0023	0	0	4.9	0.52
1408	0.0021	0.0044	0	0	8.4	1.52
1414	0.0035	0.0079	0	0	14.0	4.20
1424	0.0057	0.0137	0.0007	0	30	9
1440	0.0092	0.0230	0.0010	0	32	22
1466	0.0150	0.0380	0.0025	0	50	55
1509	0.0245	0.0623	0.0057	0	75	130
1580	0.0400	0.1025	0.0114	0	114	280
1690	0.0630	0.1650	0.0222	0	163	573
1855	0.0940	0.2600	0.033	0.017	215	1000
2077	0.127	0.386	0.000	0.073	215	1000
2333	0.146	0.532	0.000	0.092	215	1000
2575	0.137	0.670	0.000	0.083	215	1000
2765	0.108	0.778	0.000	0.054	215	1000
2889	0.071	0.849	0.000	0.017	215	1000
2960	0.0405	0.890	−0.010	0	190	200
2997	0.0211	0.910	−0.025	0	140	550
3015	0.0103	0.920	−0.020	0	100	300
3023	0.0046	0.926	−0.015	0	80	200
3027	0.0023	0.927	−0.005	0	50	100
3028	0.0003	0.927	0	0	0	0
3027	−0.0005	0.926			0	0

TABLE E-5. NUCLEAR EXCURSION CALCULATION FOR A HYPOTHETICAL CASE INVOLVING FUE

Conditions: Accident relieved by fuel expulsio
Fuel: $NaF\text{-}ZrF_4\text{-}UF_4$
Time Interval: 0.010 sec
Mean Neutron Lifetime: 4×10^{-4} sec

(a) Time, t (msec): $\Delta t = 10$	(b) k Excess, k_{ex}: $0.0004\,[(a) - (\Delta t/2)] - (j)_{n-1}/15$	(c) Period, τ (msec): $0.4/(b)$	(d) Power, P (Mw): $(d)_{n-1}\, e^{\Delta t/(c)}$	(e) Net Heat per Interval from Excursion, Q' (Btu): $[(d)_n + (d)_{n-1} - 120]\,\Delta t/2$	(f) Temperature Rise in Time Interval, $\Delta T'$ (°F): $(e)/180$	(g) Fuel Temperature Rise, ΔT_n (°F): $\sum (f)$	(h) Mean Fuel Temperature T_F (°F): $(g) + 1400$	(i) Change in F Volume in Time Interv $\Delta V'$ (ft^3): $(f) \times 5.7 \times 1$
0	0	∞	60	0	0	0	1400	0
10	0.002	200	63.3	16.5	0.09	0.09	1400	0.000057
20	0.006	66.7	73.5	84.0	0.47	0.56	1401	0.00027
30	0.00978	40.1	94.3	239	1.33	1.89	1402	0.00076
40	0.01393	26.8	136.0	551.5	3.06	4.95	1405	0.00174
50	0.01781	22.4	211	1,135	6.31	11.3	1411	0.00360
60	0.02157	18.5	362	2,265	12.6	23.9	1424	0.00718
70	0.02509	15.9	679	4,605	25.6	49.5	1450	0.0146
80	0.02812	14.2	1,373	9,660	53.7	103	1503	0.0306
90	0.03008	13.3	2,912	20,825	116	216	1619	0.0661
100	0.02967	13.5	6,108	44,500	247	466	1866	0.141
110	0.02427	16.5	11,198	85,930	477	943	2343	0.272
120	0.01014	39.4	14,433	127,555	709	1652	3052	0.404
								Fuel in core b
121	−0.0130	−30.8	10,400	12,500	70	1722	3122	0.051
122	−0.0160	−25.0	10,000	10,000	55	1775	3175	0.062
123	−0.0200	−20.0	9,500	9,500	53	1828	3228	0.073
124	−0.0245	−16.3	8,900	9,100	50	1878	3278	0.087
125	−0.030	−13.3	8,300	8,500	47	1925	3325	0.101
126	−0.036	−11.1	7,600	7,900	44	1969	3369	0.115
127	−0.043	−9.3	6,800	7,100	39	2008	3408	0.130
128	−0.051	−7.8	6,000	6,300	35	2043	3443	0.146
129	−0.060	−6.7	5,200	5,500	30	2073	3473	0.162
130	−0.070	−5.7	4,400	4,700	26	2099	3499	0.178
131	−0.081	−4.9	3,600	3,900	22	2121	3521	0.194

DEPOSITION IN THE CORE TO GIVE AN INITIAL RATE OF INCREASE IN k_{eff} OF 40%/sec

om core forced by thermal expansion

(*j*) Total Change in Fuel Volume, ΔV_n (ft^3): $(g) \times 5.7 \times 10^{-4}$	(*k*) Change in Volume Absorbed in Pressure Shell Dilation, ΔV_D (ft^3): $7.6 \times 10^{-5}[(n)_n - (n)_{n-1}]$	(*l*) Change in Volume Absorbed at Blowout Disk, ΔV_e (ft^3)	(*m*) Fuel Exit Velocity u (fps): $[(i) - (k) - (l)] \times 10^3/(\Delta t \times 0.025)$	(*n*) Velocity Pressure in 2.1-in.-dia Jet, ΔP (psi): $0.0216\ (k)^2$	(*o*) Pressure to Accelerate 30-in.-dia Inconel and Lead Disks (psi)
0	0		0	0	
0.000057	0		0.24	0	
0.00032	0		1.08	0	
0.00108	0		3.04	0	
0.0028	0		7.05	1	
0.00644	0		14.40	4	
0.0136	0.00100		25	13	
0.0282	0.0029		49	51	
0.0587	0.0087		90	165	
0.125	0.028	0	160	520	
0.266	0.036	0.049	215	1000	
0.538	0.000	0.218	120	300	
0.940	−0.068	0.448	120	300	300
s					
0.991	−0.008	0.011			230
1.053	0	0.011			230
1.126	0	0.014			300
1.213	0	0.014			300
1.314	0	0.014			300
1.429	0	0.015			320
1.559	0	0.016			330
1.705	0	0.016			330
1.87	0	0.016			330
2.05	0	0.016			330
2.24					

TABLE E-6. NUCLEAR EXCURSION CALCULATION FOR A HYPOTHETICAL CASE

Conditions: Accident relieved by boiling of
Fuel: $NaF\text{-}ZrF_4\text{-}UF_4$
Time Interval: 0.010 sec
Mean Neutron Lifetime: 4×10^{-4} sec

(a)	(b)	(c)	(d)	(e)	(f)	(g
Time, t (msec): $\Delta t = 10$	k Excess, k_{ex}: $0.0006\,[(a) - (\Delta t/2)] - (j)_{n-1}/15$	Period, τ (msec): $0.4/(b)$	Power, P (Mw): $(d)_{n-1}\, e^{\Delta t/(c)}$	Net Heat per Interval from Excursion, Q' (Btu): $[(d)_n + (d)_{n-1} - 120]\,\Delta t/2$	Temperature Rise in Time Interval, $\Delta T'$ (°F): $(e)/180$	Fu Tempe Ris ΔT_n Σ
0	0	∞	60	0	0	0
10	0.003	133	64.7	23.4	0.13	0
20	0.009	44.4	81.0	128	0.71	0
30	0.01497	26.7	118	395	2.9	3
40	0.02089	19.1	199	985	5.47	8
50	0.02668	15.0	388	2,335	13.0	21
60	0.03219	12.4	869	5,685	31.6	53
70	0.03699	10.8	2193	14,710	81.7	135
80	0.03988	10.0	5961	40,170	223.0	358
90	−0.013	−30.8	4320	50,900	283	640
100	−0.0238	−16.8	2380	32,900	183	824
110	−0.0308	−13.0	905	15,800	88	912
120	−0.0340	−11.8	388	5,880	33	945
130	−0.0353	−11.3	159	2,140	12	957
140	−0.0353	−11.2	65	520	3	960
150	−0.0358	−11.2	27	−140	−1	959

OLVING FUEL ENTERING MODERATOR COOLING PASSAGES TO GIVE AN INITIAL RATE OF 60% $\Delta k/k$ PER SECOND

n moderator passages and fuel expulsion from core forced by thermal expansion

(*b*) Mean Fuel Temperature, T_F (°F): (*g*) + 1400	(*i*) Change in Fuel Volume in Time Interval, $\Delta V'$ (ft^3): (*f*) × 5.7 × 10^{-4}	(*j*) Total Change in Fuel Volume, ΔV_n (ft^3): (*g*) × 5.7 × 10^{-4}	(*k*) Change in Volume Absorbed in Pressure Shell Dilation ΔV_D (ft^3): $7.6 \times 10^{-5}[(n)_n - (n)_{n-1}]$	(*l*) Change in Volume Absorbed at Blowout Disk, ΔV_e (ft^3)	(*m*) Fuel Exit Velocity, *u* (fps): $[(i) - (k) - (l)] \times 10^3/(\Delta t \times 0.025)$	(*n*) Velocity Pressure in 2.1-in.-dia Jet, ΔP (psi): 0.0216 (*k*)^2
1400	0	0	0	0	0	0
1400	0.000074	0.000074	0	0	0.3	0
1401	0.00040	0.00048	0	0	1.6	0
1403	0.00125	0.00173	0	0	5.0	0.540
1408	0.00312	0.00484	0	0	12.5	3.40
1422	0.00741	0.0123	0.0011	0	25	14
1452	0.0180	0.0303	0.0041	0	55	69
1535	0.0466	0.0770	0.018	0	116	300
1758	0.127	0.204	0.053	0.020	215	1000
iling Expels Fuel from Moderator						
2041	0.161	0.366	0.000	0.107	215	1000
2241	0.105	0.471	0.000	0.048	215	100
2312	0.050	0.520	−0.004	0	215	1000
2345	0.019	0.539	−0.026	0	180	600
2357	0.007	0.546	−0.030	0	100	200
2360	0.000	0.546	−0.016	0	54	60
2351	0.000	0.540	0	0	0	0

Appendix F

EFFECTS OF A NUCLEAR ACCIDENT ON REACTOR STRUCTURE

The detailed design of the reactor and related equipment has been predicated upon the use of a stress level approximately one-fifth of the stress for rupture in 1000 hr. Thus the system is one in which no burst-type of rupture will be likely to occur. The type of failure to be expected would be the result of either a fatigue crack or a leak caused by corrosion. In either case, the failure would develop slowly so that there would be ample indication of the character of the trouble before anything serious developed. It should be noted, however, that while this philosophy has been applied to the types of accident and hazards to be expected, no advantage of this design basis has been taken in consideration of extreme accidents. Such accidents could take place only if a burst-type of rupture occurred, and such ruptures have been presumed, even though no reasonable mechanism for causing them has been envisioned.

In addition, two major tenets of the design philosophy have been that the pressures throughout the systems should be kept low, particularly in the hot zones, and that all structure should be cooled to a temperature approximately equal to or below that of the secondary coolant leaving the heat exchanger. Great care was exercised in establishing the proportions of the designs presented in Tables F.1 and F.2 to satisfy these conditions. The stress values calculated for the various stations in a typical design are indicated in Fig. 2.1. The stresses in the structural parts have been kept to a minimum and the ability of the structure to withstand these stresses has been made as great as practicable. Thermal stresses are not indicated since they will be indeterminate, and it is felt that they will, to a large degree, anneal out at operating temperatures.

If a very severe pressure surge is assumed to occur as a consequence of a nuclear accident, it is possible to envision fairly well the sequence of events that would lead to a failure in the reactor structure. The first consideration in any such analysis is the strength of the material of the structure. Table F.3 presents strength data for Inconel, the structural material presently being considered. Note that the yield point and ultimate tensile strength are much higher than the 1000-hr stress rupture limit. Note also that the percent elongation is substantial, and thus much plastic distortion would take place before rupture would occur. Since the structure incorporates substantial stress concentrations, the local yielding may be substantial, but the volumetric change in the pressure shell will be small.

TABLE F.1. KEY DIMENSIONAL DATA FOR THE ART PUMP-HEAT EXCHANGER-PRESSURE SHELL ASSEMBLY

Core	
Core diameter (in.)	21
Island outer diameter (in.)	10.75
Core inlet outer diameter (in.)	11
Core inlet inner diameter (in.)	7
Core inlet area (in.^{2})	56.5
Fuel System	
Fuel volume in core (ft^3) (30 in. length)	2.96
Fuel volume in inlet and outlet ducts (ft^3)	0.25
Fuel volume in heat exchanger (ft^3)	2.23
Fuel volume in pump volutes (ft^3)	0.2
Total fuel volume in main circuit (ft^3)	5.64
Fuel expansion tank volume (ft^3) (8%)	0.45
Expansion tank diameter (in.)	14
Expansion tank height (in.)	6
Fuel Pumps	
Centerline to centerline spacing	20
Volute chamber width (in.)	12
Volute chamber length (in.)	32
Volute chamber height (in.)	1.1
Impeller speed (rpm)	2850
Estimated impeller weight (lb)	11
Critical speed (rpm)	6000
Sodium Pumps	
Speed (rpm)	4300
Impeller diameter (in.)	5.75
Impeller inlet diameter (in.)	2.75
Impeller discharge height (in.)	0.50
Sodium System	
Expansion tank volume (ft^3) (10%)	0.11
Sodium in beryllium passages (ft^3)	0.18
Sodium in pressure shell (ft^3)	0.55
Sodium in pump and heat exchanger (ft^3)	0.10
Sodium in return from moderator (ft^2)	0.20
Total sodium volume (ft^3)	1.18
Main Heat Exchanger	
Volume (ft^3)	6.3
Number of tube bundles	24
Number of tubes per bundle (11 x 12)	132
Total number of tubes	3168
Latitude of header centerline (deg)	45
Sodium-to-NaK Heat Exchangers	
Number of tube bundles	2
Number of tubes per bundle	300
Moderator Regions	
Volume of beryllium plus fuel (ft^3)	27.2
Volume of beryllium only (ft^3)	24.0
Cooling passage diameter (in.)	0.187
Number of passages in island	46
Number of passages in reflector	232

TABLE F.2. DIMENSIONS OF ART DETAIL PARTS

Equatorial Diameters (in.)

Island	
Control Rod Thimble, ID	.50
Control Rod Thimble, OD	.58
Be Island, OD	10.25
Inner Core Shell, ID	10.50
Inner Core Shell, OD	10.75
Inner Core Shell Thickness	0.125
Reflector	
Outer Core Shell, ID	21.00
Outer Core Shell, OD	21.25
Outer Core Shell Thickness	0.125
Be Reflector, ID	21.37
Be Reflector, OD	44.20
Reflector Inner Boron Jacket, ID	44.26
Reflector Inner Boron Jacket, OD	44.31
Boron Layer, ID	44.31
Boron Layer, OD	44.56
Reflector Outer Boron Jacket, ID	44.56
Reflector Outer Boron Jacket, OD	44.61
Reflector Shell, ID	44.75
Reflector Shell, OD	45.00
Pressure Shell	
Boron Jacket, ID	49.61
Boron Jacket, OD	49.74
Boron Layer, ID	49.74
Boron Layer, OD	49.99
Liner, ID	49.99
Liner, OD	51.49
Pressure Shell, ID	51.74
Pressure Shell, OD	53.74
Pressure Shell Thickness	1.00

Vertical Distance Above Equator (in.)

Floor of Fuel Pump Inlet Passage	18.5
Bottom of Lower Deck	19.5
Top of Lower Deck	20.0
Bottom of Upper Deck	23.0
Top of Upper Deck	23.5
Top of Fuel Pump Mounting Flange	49.0
Top of Na Pump Mounting Flange	51.0

Fuel Pump Assembly Dimensions (in.)

Shaft	
Thrust Bearing Journal, OD	1.3785
OD Between Bearings	1.968
Lower Bearing Journal, OD	1.9965
Seal Washer Journal, OD	1.875
OD Below Seal	1.859

Thread for Impeller Retaining Nut	1.687 - 12 N.S.
Thrust Bearing Height from Equator	47.812
Lower Bearing Height from Equator	35.812
Distance Between Bearings	11.00
Distance Between Thrust Bearing and Impeller Locating Shoulder	25.437
Over-all Shaft Length	32.437

Impeller

OD	5.75
Inlet, ID	3.5
Discharge Passage Height	1.0
No. of Vanes	5
Axial Distance from Top of Discharge Passage:	
Top of Centrifuge	2.25
Shaft Locating Shoulder	.87
Inlet Face	2.00

Pump Body

OD of Locating Journal Outside Lower Bearing	5.935
OD of Flange at Top	9.37
No. of 1/2 - 20 Machine Screws in Flange	8

Vertical Distances from Lower Face of Mounting Flange

Bottom of Thrust Bearing	.750
Face of Shaft Seal Washer	13.312
Bellows Seal Mounting Pad	15.344
Lower Face of Boron Jacket	20.750
Top of Impeller Discharge Passage	26.437

Main Heat Exchanger (in.)

Tube Centerline to Centerline Spacing	0.208
Tube OD	0.1875
Tube ID	0.1375
Tube Wall Thickness	0.025
Tube Spacer Thickness	0.020
Mean Tube Length	72
Radius of Equatorial Reverse Bend	1.5
Inlet and Outlet Pipe ID	1.65
Inlet and Outlet Pipe OD	1.95
Header Sheet Thickness	0.25
Header Sheet Inner Radius	2.25

Moderator Circuit Heat Exchanger (in.)

Tube Centerline to Centerline Spacing	0.208
Tube OD	0.1875
Tube ID	0.1375
Tube Wall Thickness	0.025
Tube Spacer Thickness	0.020
Mean Tube Length	28

TABLE F.3. STRENGTH DATA FOR INCONEL TESTED IN A FLUORIDE MIXTURE

Temperature, °F	1200	1500
Ultimate tensile strength, psi	65,000	21,000
Yield point, psi	26,500	13,000
Stress for rupture in 1000 hr, psi	17,000	2100
Elongation, %	39	55

Stress Calculations

The principal element in the reactor structure is the main pressure shell. It is a spherical vessel 1 in. thick with an internal diameter of 52.5 in. During normal operating conditions the temperature of the pressure shell will be 1200°F. The stress in this shell for an internal pressure P is given by

$$S = \frac{Pr}{2t} = \frac{P \times 25.7}{2} = 12.9\ P\ .$$

The stress, S, and pressure, P, are in psi and the radius, r, and the shell thickness, t, are in inches. In re-examining the stress expression and Table F.3 it is evident that the main pressure shell would begin to yield at a pressure of about 1000 psi at stress concentrations such as those which will be present in the ligament between the heat exchanger inlet and outlet header pipes. It would be expected that progressive local yielding would continue to take place in these regions until failure occurred; some increase in pressure would occur in the process. Another major weak area would be the flat structural region at the top of the shell where the pumps and expansion tanks are to be located. While final detailed designs are not yet available, it does seem certain that local yielding would begin in this region at a pressure of the order of 1000 psi. It also seems likely that one result of this yielding would be that the pump casings would distort and cause the impellers to seize in the casings and the pumps to stop. Further distortion would tend to make this flat region spherical until high local stress concentrations had produced large amounts of yielding to the point where failure would occur.

It is instructive to examine one such stress concentration that is reasonably well defined, namely, that in the south heat exchanger outlet belt. This belt has a centerline diameter of 36.4 in. The total cross sectional area in the ligaments in the main pressure shell would be 60 in.^{2}, and this would be augmented by approximately 44 in.^{2} of reinforcing patches welded to the pressure shell between the heat exchanger outlet pipes.

The shear stress in this region would then be

$$S = \frac{Pr}{2t} = \frac{P\frac{d}{2}}{2\frac{A}{\pi d}} = \frac{P\pi d^2}{4A}$$

$$= \frac{P \times 0.786(36.4)^2}{60 + 44} = 10{,}000 \text{ psi} .$$

This would give a tensile strength of 14,100 psi, which represents an average stress in the ligaments. The peak value around the edge of the holes would be approximately twice as great so that yielding would begin there. The stress would then distribute itself uniformly across the ligament, and further plastic deformation would follow until a tear would begin and rupture would finally take place.

The stresses in the outer core shell are of particular interest because expansion of that core shell would increase the fuel volume in the reactor and as a consequence give an increase in reactivity. If the core shell were not supported by the moderator, the stress in the shell would be given by

$$S = \frac{Pr}{1.6t} = \frac{P \times 10.5}{1.6 \times 0.125} = 52.5 P .$$

Note that a factor of 1.6 instead of 2 was used in the denominator because the radius of curvature of the core shell at the equator in the vertical plane is greater than it is in the horizontal plane. This core shell is actually supported by flattened Inconel wires inserted between the core shell and the beryllium reflector and spaced in rows on 1-in. centers. These strips are used primarily to maintain the spacing between the reflector and the shell and hence to ensure the proper coolant flow passage opening. It is interesting to determine the radius of curvature of the shell when deflected between the supports by a pressure that would give the same stress in the core shell as would prevail in the main pressure shell. If a simplified cylindrical geometry is assumed, the relation for that case is

$$S = \frac{P \times 25.7}{2 \times 1} = \frac{Pr}{0.125} ,$$

from which a radius of 1.6 in. is obtained. Thus the core shell would deflect to the point where it would have a radius of curvature of 1.6 in. between supports. The effect on the core volume of this distortion coupled with that of the inner core shell would be relatively small, and would give a volume increase of about 1% for a 1000-psi pressure. The stresses in the inner core shell would be similar to, but lower than, those in the outer core shell because its diameter would be roughly one-half that of the outer core shell.

The possibility of buckling and collapse of the tubes in the main heat exchanger has also been considered. The stress in the tube walls from an external pressure is

$$S = \frac{Pr}{t} = \frac{P\ 0.094}{0.025} \cong 4P.$$

Note that this stress is very much less than that in the main pressure shell so that it is very unlikely that the heat exchanger tubes would collapse in the event of a pressure surge.

The force required to blow the pump body out of the pump casing is of interest. If it is assumed that the main pressure shell would begin to yield locally under a pressure of 1000 psi, the force acting on the pump body may be calculated in the following way:

$$\text{Pump body blow-out force} = 1000 \times 0.786 \times 5.75^2 = 26{,}000 \text{ lb.}$$

The cross-sectional area of the pump body attaching screws is over 2 in.^{2}, and since this is the weakest point in the system restraining the pump body from blow-out and since it is in a cold region where the tensile strength of the material is of the order of 90,000 psi, it is clear that this does not represent a weak point in the system. While the thrust bearing at the top of the pump impeller shaft might fail and allow the impeller shaft assembly to move upward several inches, the pump impeller itself would keep the shaft from moving on out through the pump body. If this failure took place there would be free communication from the reactor core through the pump impeller bore into the fuel expansion tank.

The volumetric stiffness of the main pressure shell is of interest from the standpoint of determining the pressure at various stages during a nuclear accident. The change in volume, ΔV, in cubic feet, is given as follows:

$$\Delta V = \frac{\Delta r\ 4\pi r^2}{1728},$$

where

$$\Delta r = \frac{rS}{E} = \frac{25.7 \times 12.9\ P}{21 \times 10^6}.$$

Thus

$$\Delta V = 7.6 \times 10^{-5}\ P,$$

where Δr is in inches and represents the increase in diameter of the inside of the pressure shell. Note that a modulus of elasticity, E, of 21×10^6 was used since the pressure shell temperature should be a little below 1200°F. From this it follows that the increase in volume inside the pressure shell for a 1000-psi pressure surge would be 0.076 ft^3. It should be noted that the pressure shell in some respects would be stiffer than it is assumed to be here because of the presence of the liner in the heat exchanger region and at the south end. On the other hand, the system would

be less stiff because of the presence of flat regions in the pump and expansion tank region. It seems likely that a rough approximation to the stiffness of the system has been given.

Destructiveness of Pressure Shell Fragments

The velocity of the fluid escaping through a crack if rupture should occur at 1000 psi is of interest. This can be computed as follows:

$$V = \sqrt{2gH} = \sqrt{64.4 \times \frac{1000 \times 144}{200}} = \sqrt{46,300} = 215 \text{ fps},$$

where H is the pressure (in lb/ft^2) divided by the density of the fluid (in lb/ft^3). This velocity of 215 fps is certainly a reasonable one and should not give any particular trouble. It is also important because it represents the maximum possible velocity of a fragment that might be broken out of the pressure shell in the event of a hydrostatic rupture. It is interesting to estimate the maximum kinetic energy that could exist in an escaping fragment for this case. The work or energy put into the fragment in the rupture process should equal double the area under the PV diagram for the pressure at rupture and the volumetric increase inside the pressure shell up to the rupture pressure. This triangular PV diagram area should be multiplied by a factor of two to allow for the elastic overshoot or rebound of the shell accompanying the pressure relief:

$$\text{Work} = P\Delta V = 144,000 \times 0.076 = 10,900 \text{ ft-lb}$$

It is also of interest to calculate the weight, W, of the minimum fragment required to carry off all the rupture energy. This is

$$\text{K. E.} = 10,900 = \frac{W}{2 \times 32.2} (215)^2$$

$$W = 15 \text{ lb.}$$

Thus it appears that the most destructive fragment from a pressure shell failure at 1000-psi pressure would be a 15-lb fragment that would leave the reactor at a velocity of 215 fps with a kinetic energy of about 11,000 ft-lb. If this fragment were of 1-in.-thick Inconel, its area would be 260 $in.^2$. If it were to strike the lead shield and all of its momentum were imparted to a lead fragment of the same area having a thickness of 7 in., the resulting velocity of the lead fragment could not exceed 20 fps if no allowance were made for the energy absorbed in rupturing the lead.

In re-examining the above analysis it appears that a number of modes of failure might be encountered. If no special provision were made to relieve the pressure it seems likely that the north head would deflect plastically to the point where the pump impellers would seize in their casings. Thus whether the increase in volume associated with this deflection would be sufficient to relieve the pressure associated with the nuclear reaction would depend upon the particular accident. Certainly, if carried far enough, rupture would occur. If rupture did occur it seems unlikely that

a large fragment or, for that matter a small fragment, would be blown out of the wall of the main pressure shell. Rather it seems likely that the wall would split and open up to relieve the excessive pressure. It seems most likely that this splitting would occur in the ligaments between the heat exchanger outlet tubes at the south end of the reactor. In that event about a 3/4-in. stretch in these ligaments would probably occur prior to failure.

Pressure Relief Mechanisms

Provision can be made in the design to relieve a violent pressure surge from a nuclear accident in a number of ways. It would be possible, for example, to put a poppet valve in the fuel region just ahead of the fuel pump inlet so that it would open and vent to the expansion tank. The utility of such a move is doubtful because the volume available in the expansion tank is probably not quite adequate to take care of an extreme nuclear accident. While an overflow will be connected from the top of the expansion tank to the emergency fuel dump tank, and this line will be kept heated at all times, its diameter would have to be at least 4 in. if it were to be effective in relieving a pressure surge in a nuclear accident. A second provision for pressure relief could take the form of a frangible diaphragm. If this frangible diaphragm were placed at the bottom of the reactor it would be in a high-pressure region and would be subjected to a pressure of approximately 70 psi under normal operating conditions. Thus it would have to be designed to rupture at a pressure of at least 400 psi because the ratio of the ultimate tensile strength to the 1000-hr rupture strength is approximately 4 at 1200°F. Thus, if a factor of safety of 4 were used in the design of the disk, under normal operating conditions the pressure at that point would have to be roughly 16 times as great to produce failure, or about 1000 psi. If, on the other hand, the diaphragm were placed at, or close to, the fuel pump inlet where the pressure would normally be, perhaps 5 psi, it could be designed to rupture at a pressure of 60 psi and still have a factor of safety of 12, stress-wise, for normal operating conditions. Yet another device that might be used would be to employ a weak seam in the attachment of the whole north head or some part thereof. This again would have the disadvantage that it would operate under high stress for normal operating conditions if it were designed for rupture at a pressure below 1000 psi.

If the pressure shell or a frangible diaphragm ruptures, there must be space for expansion of the fuel or the lead shell will also be ruptured. Since there is a 1/2-in.-thick thermal insulation zone between the pressure shell and the lead region and since this zone is only partially filled with reflective insulation there is available an expansion volume of 1.3 ft^3 immediately outside the reactor pressure shell. Additional volume could be provided readily with little penalty in shield weight by leaving voids in various places in the irregular region around the pumps and header tank. Shielding placed in the re-entrant corners is relatively ineffective in this region and could be omitted. In this way an explosion volume of perhaps 0.5 ft^3 could be readily arranged. Yet another way in which expansion volume could be provided would be to place a blister on one side or the top or the bottom of the pressure shell. The side or bottom of the pressure shell is particularly attractive from this standpoint because a drain line

from the blister to the fuel dump tank could be easily installed. Of course, a large diameter pipe could be extended directly out through the shield. However, since a pipe of at least 4 or 5 in. in diameter would be required, this would constitute a serious gap in the shield which would be difficult to block off with a shield patch.

While it is hard to see how it could occur, an extreme power excursion relieved by boiling at the peak of the power curve is of interest. At first glance, it appears that such an accident might lead to the ejection of pressure shell fragments that could pierce the wall of the inner tank. In re-examining the stress analysis results, together with the curves of Fig. E.1 and the data in Tables E.2 to E.6, two points are evident. The first is that the shell will probably rupture locally under an internal pressure of between 1000 and 2000 psi and will certainly rupture generally at a pressure not to exceed 4000 psi. The second point is that if a small opening having a diameter of say 4 in. appears, the rate of ejection will not be high enough to keep the fluid pressure from rising so much that either the hole would tear open further or a second failure would occur. In either case it would not be possible for all of the energy to be concentrated in a single small fragment travelling at a very high velocity. In fact, from Bernoulli's equation, the limiting velocity for the fluid column (or any fragment) if the peak pressure were 4000 psi would be:

$$V = \sqrt{2gH} = 430 \text{ fps.}$$

The penetrating power of projectiles varies with their shape, hardness and strength, mass, and velocity. While an ogival shape is the most penetrating, a fragment of such a shape seems quite out of the question. A sphere has greater penetrating power than any irregular shape likely to be ejected from the pressure shell. The old Krupp formula for the penetration of wrought iron plate by cast iron cannon balls appears to be roughly applicable to this case, i.e.,

$$V = 2000 \left(\frac{t}{d}\right)^{2/3}$$

where V is in fps and t and d are in inches. The velocity of a 6-in. sphere required to penetrate 1/2-in. plate is then

$$V = 2000 \left(\frac{0.5}{6}\right)^{2/3} = 380 \text{ fps.}$$

Approaching the problem from a different standpoint and examining the pressure shell structure carefully, it appears most likely that a circular section 36.4 in. in diameter would be sheared out through the heat exchanger outlet belt. By using the numerical method outlined earlier and by assuming that there was no loss in reactivity until boiling began to expel fuel from the core and that all of the pressure relief was obtained by shearing out the bottom of the pressure shell, the results for a very severe nuclear power surge caused by increasing k_{eff} at a rate of 40% per second were obtained, as shown in Fig. E.1. The fragment would be accelerated until it had passed nearly through the lead region. It was assumed in the

calculations that yielding in the Inconel shell took place until it had moved through the clearance for thermal insulation and contacted the lead gamma shield, at which point rupture of the Inconel occurred. No allowance was made for lateral escape of fluid into the space between the lead and the Inconel shells, nor was allowance made for the energy absorbed in shearing the lead. Note that the velocity of the Inconel and lead fragments at the end of the expansion is only about 40 fps. While the failure might start first on one side so that the disk might tilt during the process of failure, this would probably result in opening a flap in the pressure shell without tearing the flap loose from the shell and ejecting it as a projectile. The worst case would occur if the flap were ejected and tilted in flight so that it would strike the tank bottom edgewise. The velocity required for such a projectile to pierce the tank wall can be estimated from the relation for the energy required to punch holes in plate, i.e.,

$$\frac{WV^2}{2g} = t^2Bs$$

If the Inconel disk is 1 in. thick, its weight, W, will be about 350 lb, and the periphery, B, of the hole it will punch will be about 76 in. The ultimate tensile strength, s, of the steel plate should be about 60,000 psi. The velocity, V, required to pierce the tank wall having a thickness t = 0.5 in. would then be

$$V = \sqrt{\frac{2t^2Bsg}{W}} = \sqrt{\frac{0.5 \times 76 \times 60{,}000 \times 32.2}{12 \times 350}}$$

$$= \sqrt{63{,}000} = 251 \text{ fps.}$$

This velocity is comparable to the 380 fps found for the 6-in. sphere from the Krupp formula. While the mass of the lead piece sheared out of the gamma shield would be large, it would be soft so that most of its energy upon impact would go into deforming itself rather than into piercing the tank wall.

A critical examination of the results of the numerical analyses discloses that the reactor should not be damaged by a nuclear accident of the type shown in Fig. E.2 if the initial rate of increase in k_{eff} did not exceed 10% $\Delta k/k$ per second. More severe accidents that did not result in boiling of the fuel would probably cause distortion in the pressure shell that would interfere with pump operation and possibly cause leaks. While a light frangible diaphragm could be used to relieve such accidents so that the pressure shell structure would not be damaged, any such incident would involve so severe a temperature overshoot that further operation of the reactor would not seem to be prudent. Thus there seems to be little point in making the design compromises necessary for a light frangible diaphragm. This is particularly true in view of the very remote probability of accidents in which it would be of benefit and the much greater likelihood of troubles that it might give in the course of routine operation. For this reason the analysis in this report and the curves in Figs. E.1 and E.2 have been prepared by presuming that up to pressures of 1000 psi the only

avenue of escape of fuel from the core circuit is through the clearances between the pump impellers and their casings and through the centrifuge discharge holes into the fuel expansion tank. It should also be noted that the worst case in Fig. E.2 that appears, on this basis, to leave the reactor undamaged is that for an initial rate of increase in k_{eff} of 10% $\Delta k/k$ per second for which the volume of fuel expelled is about equal to the free space normally available in the expansion tank. For cases tending to give pressures above 1000 psi, it was deemed best to incorporated a circular groove in the bottom of the pressure shell so that the pressure would be relieved by failure in that region. The fuel would be ejected downward and a minimum of damage to the system would result.

Appendix G

EXPOSURE HAZARD CALCULATIONS[1]

Criteria

The calculations of the exposure hazard were based on a total internal exposure of not more than 25 rem to any internal organs (bone, thyroid, lungs, G. I. tract, kidneys) over the lifetime of the people exposed. Since the general population is being considered, the lifetime is taken to be 70 years. The group of people to be protected includes children, the pregnant, those especially radiosensitive, those with large previous exposure records, and those occupationally exposed. If D is the exposure rate to the organ, then

$$25 \text{ rem} \geq \int_{t=0}^{t=70 \text{ y}} D \, dt \quad ,$$

where time, t, begins with the intake of the isotope mixture. The intake may be either by inhalation or by ingestion.

Basic Formulas

The exposure rate, D, to an organ is the sum of the exposure rate from material a, D_a, the exposure rate from material b, D_b, etc. Therefore,

$$D = \sum_i D_i \text{ (rem/day)} \quad ,$$

where D_i is the exposure rate to a body organ resulting from its content of isotope i. The exposure rate from isotope i at time t then is

$$\begin{aligned} D_i^t = \Big\{ & Q_i \text{ } (\mu c) \times 3.7 \times 10^4 \text{ (dis/sec}\cdot\mu c) \times 8.64 \times 10^4 \text{ (sec/day)} \\ & \times \sum_i E \text{ (Mev/dis)} \big[\text{RBE}\big] N \text{ (rem/rad)} \\ & \times 1.6 \times 10^{-6} \text{ (erg/Mev)} \Big\} e^{-\lambda_i t} \Big/ \big[m \text{ (g)} \times 100 \text{ (erg/g}\cdot\text{rad)} \big] \\ = 51.2 & \frac{Q_i \sum_i E(\text{RBE})N}{m} e^{-\lambda_i t} \text{ (rem/day)} \quad , \end{aligned}$$

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1. This section prepared by T. J. Burnett, ORNL Health Physics Division.

at time, t, where

Q_i = amount of isotope i in organ (μc),

m = mass of organ containing Q_i (g),

$\sum_i E(RBE)N$ = energy, E, dissipated in organ of mass m from each disintegration of isotope i, weighted for biological effectiveness, RBE, and nonuniformity of distribution, N,

(RBE = 1 for γ, β^-, x, and e^-

= 10 for α

= 20 for recoils

N = 1 for γ and x

= 5 for α, β^-, β^+, e^-, and atom recoils in bone

= 1 for all body organs except bone),

λ_i = elimination rate constant for isotope i from the body organ
$= 0.693/T_i$, where

T_i = effective half life of isotope i from body organ
$= \frac{T_b T_r}{T_b + T_r}$, where T_b is the biological half life and T_r is the radioactive half life.

(In the expression of the exposure rate the unit "rad" is the dose from deposition of 100 ergs per gram of tissue).

The total exposure to the body organ from isotope i over any period t following a single intake is

$$D_i^t = \int_{t=0}^{t} D_i^t \, dt = D_i^o \frac{T_i}{0.693} \left(1 - e^{-\lambda t}\right),$$

where D_i^o is the initial dose rate from isotope i over a period of 70 years (= 25,568 days); thus

$$D_i^{70y} = D_i^o \frac{T_i}{0.693} \left[1 - e^{-\lambda t \ (=70y \text{ in days})}\right] .$$

The total exposure to the body organ from a mixture of isotopes is then $\sum_i D_i^t$, in which the amounts of the components i which reach the organ following intake are considered. For a mixture, the values of Q_i will depend on the composition of the mixture and the uptake of components to the organ.

Method

A unit intake (inhalation) of 1 μc of the mixture of interest is considered. The total dose of each component of this mixture is calculated by using its fraction of the total of the mixture considered. The component doses are then summed for the organs affected. Based on the sum of these total doses, the number of microcuries of the mixture which can be inhaled to give a total dose to any organ of 25 rem in 70 years is calculated. This number of microcuries is the total permissible inhalation intake for a single exposure to the mixture considered $\left(MPI^{25\text{ rem, }70\text{ y}}_{\text{single exposure}}\right)$.

Meteorological calculations of inhalation exposure (based on Sutton's equations) yield total integrated dose values in curie-seconds per cubic meter as the integral of concentration with time

$$TID = \int x\ (c/m^3)\ dt\ (sec) = curies \cdot sec/m^3$$

The TID times the breathing rate in m^3/sec gives the inhalation intake in curies.

The maximum permissible total integrated dose in curies·sec/m^3 is

$$MPTID = \frac{MPI_{se}^{25/70}}{BR},$$

where BR is the breathing rate. Various values for breathing rates may be considered and are appropriate to the conditions of rest, exercise, excitement, etc., of the individuals exposed.

For these calculations the breathing rate has been given a value of 30 liters per minute, which corresponds to moderate exercise and possible excitement (since the application is to instances of a reactor accident):

$$BR = \frac{30\ (\ell/min) \times 10^{-3}\ (m^3/\ell)}{60\ (sec/min)} = 5 \times 10^{-4}\ m^3/sec$$

$$MPTID = \frac{MPI_{se}^{25/70}\ (curies)}{5 \times 10^{-4}} = 2 \times 10^3 \times MPI_{se}^{25/70}\ curies \cdot sec/m^3.$$

If a unit inhalation intake of 1 μc is considered,

$$Q_i = \%_i \times f_{a_i} \ \mu c \text{ in organ per inhaled } \mu c,$$

where $\%_i$ is the fraction of the mixture that is isotope i and f_{a_i} is the fraction of the inhaled isotope i that is retained in the critical organ. The critical organ is that organ of the body which receives the isotope that results in the greatest to the body. In most cases it is that organ which receives the greatest damage. However, some organs are less essential to the well-being of the entire body. Usually the critical organ is that which receives the greatest concentration, but there is considerable variation in sensitivity. The critical organ depends also on the mode of intake and may change with time after intake.

Calculational Procedure

The components of a given mixture of mixed fission products are grouped by the body organs affected. A tabulation of $\%_i$, f_{a_i}, and $\sum_i E(RBE)N$ values is then made and the corresponding values of D_i^o are calculated by using the m of the organ. These can be summed to give the initial dose rates to the organ.

The values of D_i are then tabulated, together with those for T_i. By combining these with values of $(1 - e^{-\lambda_i t})$ for various times, t, following intake, the total doses D_i^t are found. The values of D_i^t are then summed, and the totals are used to calculate $MPI_{se}^{25/70}$.

Values of $\%_i$ will vary with the mixture of isotopes considered, which in turn depends on the time of reactor operation. This is also sensitive to the choice of isotopes for the mixture to be considered.

The components of the selected mixture were chosen because of their known hazard and for half lives generally in the range of 20 hr to 20 years. Shorter-lived isotopes would have D_i^t values small enough to ignore, and longer-lived isotopes would have small $\%_i$ values for the operational times of interest with high-power-density reactors where the reactor is operated with a max. irradiation time of the order of 1000 hr. Buildup of longer lived materials is small for these times. Parent-daughter relations also influenced the choice of some isotopes and their inclusion. Since the composition of the mixture chosen will vary with time following a reactor incident because of the differing decay rates of its components, there would be corresponding different $MPI_{se}^{25/70}$ values at different points sufficiently far downwind for the airborne transit time to introduce these decay differences. As a first approximation the decay differences can be ignored, since the mixture chosen undergoes small decay for the relatively short times of interest for most reactor accidents for which the radius of hazard is small.

For this first approximation, the $MPI_{se}^{25/70}$ based on initial concentration can be used. It is probable that the effect of decay may be compensated by increases of the relative $\%_i$ values and, since the longer lived materials are more hazardous, the net effect could be smaller $MPI_{se}^{25/70}$ values at more distant points where decay would be significant.

The $\sum_i E(RBE)N$ values used are either those given by the International Committee on Radiological Protection (ICRP) or are those calculated as prescribed in the ICRP Internal Dose Subcommittee Report of December 7, 1953 (K. Z. Morgan, chairman). The prescribed formula is

$$\sum E(RBE)N = \sum_{j,k,m} \left[f_{\gamma_j} E_{\gamma_j} (1-e^{-\sigma_j x}) + 0.33 f_{\beta_k} E_{\beta_k} \left(1 - \frac{z^{1/2}}{43}\right) \left(1 + \frac{E_{\beta_k}}{4}^{1/2}\right) N_{\beta_k} + f_{e^-_m} E_{e^-_m} N_{e^-_m} \right],$$

which considers the decay scheme of the isotope involved. In this formula RBE and N are as previously defined and

f_{γ_j} = fraction of the disintegrations of the j^{th} type that result in the emission of a gamma or x photon of energy E_{γ_j},

f_{β_k} = fraction of the disintegrations of the k^{th} type that result in the emission of a beta ray of maximum energy E_{β_k},

$f_{e^-_m}$ = fraction of the disintegrations of the m^{th} type that result in the emission of a conversion electron of energy $E_{e^-_m}$,

σ_j = total coefficient of absorption minus Compton scattering coefficient in cm^{-1} for tissue for photons of energy E_{γ_j} (from ORNL-421, Dwg. 8117).

x = effective radius (in cm) of the critical body organ containing the radioisotopes; values of x are given in Table 2 of the ICRP report cited,

z = atomic number of the radioisotope emitting the beta ray.

When appropriate, the parent-daughter combinations are considered together; f_{a_i} values are taken from the ICRP report cited, and T_i values are either taken from the ICRP report or are calculated by using the T_b values (from the ICRP report) and the T_r values (from Hollander, Perlman, and Seaborg, J. Mod. Phys. Apr. 1953, which was also used as the chief source of decay schemes and energies).

For the purpose of these calculations a mixture of about thirty selected isotopes was considered. The relative compositions were based on an irradiation time of 39 days ($\sim$ 940 hr), and decay from the time of release to the time of inhalation was ignored. The curies present at release time were calculated to be

$$c_i^o = 90.8 \times 10^4 \times y_i s_i \text{ curies/Mw of power level,}$$

where

c_i^o = curies of isotope i at time o (release time),

y_i = yield of isotope i from fission

s_i = saturation factor for isotope i, taking into account its growth,

$$90.8 \times 10^4 = \frac{3.36 \times 10^{16} \text{ (dis/sec/Mw)}}{3.7 \times 10^{10} \text{ (dis/sec/curie)}}$$

The value 3.36×10^{16} dis/sec/Mw was obtained by using 186 Mev/fission as given by Wigner in ORNL-1638, Project Hope (p. 16). The value given by Wigner is in agreement with the Bulk Shielding Facility value of 190 $\pm$ 5 Mev/fission determined by J. L. Meem and reported in ORNL-1537. Thus,

$$\frac{10^7 \text{ (ergs/sec/watt)} \times 10^6 \text{ (watts/Mw)}}{1.6 \times 10^{-6} \text{ (erg/Mev)} \times 186 \text{ (Mev/fission)}} = 3.36 \times 10^{16} \text{ fission/sec/Mw.}$$

The use of the more traditional 3.1×10^{16} value would only introduce a scale factor and not alter the relative $\%_i$ values.

The y_i values used are taken from Glendenen et al., PR 84, 4, 860, and ANL-OCS-383 (Letter Glendenen and Steinberg to R. W. Stoughton, Dec. 11, 1953) with interpolated values chosen to give $\sum_i y_i = 2.0457$.

The results of the calculations are presented in Tables G.1 through G.4. Table G.1 gives the fission yields used; Table G.2 presents the selected isotope mixture; Table G.3 lists the initial dose rates; and Table G.4 gives the total doses in 70 years.

From Table G.4 it is seen that:

1. The exposure to the bone is controlling.
2. The $MPI_{se}^{25/70} = \frac{25000}{53.1} = 717$ μc (4.7×10^{-4} curies).
3. For this mixture (39 days), the maximum permissible total integrated dose = 1.44 curie·sec/m^3.

4. There are six major contributors to the dose to the bone. These are tabulated separately in Table G.5.

5. These six isotopes contribute 95.6% of the total 70-year dose to the bone.

6. The total curies of the mixture of these six isotopes alone is less than the total of the 30 isotopes by a factor of 7. The total dose (corrected by the 95.6% factor) is greater by this same factor.

7. No significant decay occurs with these isotopes for times of interest.

8. The effect of lesser irradiation times is significant.

TABLE G.1. FISSION YIELDS

Chain No.	y_i (%)	Chain No.	y_i (%)
79	0.03	125	0.03
80	0.07	126	0.08
81	0.12	127	0.20
82	0.22	128	0.51
83	0.468	129	1.00
84	0.872	130	1.90
85	0.253	131	2.97
86	1.67	132	4.45
87	2.76	133	6.62
88	3.78	134	7.81
89	5.15	135	6.56
90	6.10	136	6.42
91	6.24	137	6.19
92	6.45	138	6.12
93	6.87	139	6.09
94	6.85	140	6.07
95	6.77	141	5.90
96	6.75	142	5.43
97	6.56	143	6.10
98	6.24	144	5.76
99	5.90	145	4.20
100	6.80	146	3.40
101	5.29	147	2.45
102	4.37	148	1.85
103	3.35	149	1.25
104	1.93	150	0.75
105	1.00	151	0.40
106	0.40	152	0.12
107	0.11	153	0.06
108	0.06	154	0.02
109	0.02		
	103.45		101.12

TABLE G.2. SELECTED ISOTOPE MIXTURE*

Isotope	s_i (39 days)	y_i	t=0 curies x 10^3/Mw	$\%_i{}^{39}_{30}$
Sr^{89}	0.3995	0.0515	18.681	2.1168
Sr^{90}-Y^{90}	0.00371	.0610	0.2055	0.0233
Sr^{91}-Y^{91m}	1.0	.0624	56.659	6.420
Y^{91}	0.358	.0624	20.284	2.2984
Zr^{95}	0.340	.0677	20.900	2.368
Nb^{95m} 1%	0.0034**	.0677	0.209	0.024
Nb^{95}	0.1066	.0677	6.553	0.743
Zr^{97}-Nb^{97m}-Nb^{97}	1.0	.0656	59.565	6.749
Mo^{99}-Te^{99m}	1.0	.0590	53.572	6.070
Ru^{103}-Rh^{103m}	0.493	.0335	14.996	1.699
Rh^{105}	1.0	.0100	9.080	1.029
Sb^{127}	1.0	.0020	1.816	0.206
Te^{127m} 16%	0.0335**	.0020	0.061	0.007
Te^{127}	0.874	.0020	1.587	0.180
Te^{129m} 19%	0.1053**	.0100	0.956	0.108
Te^{129}	0.9153	.0100	8.311	0.942
Te^{131m}-Te^{131}	1.0	.0297	26.968	3.056
I^{131}	0.966	.2097	26.051	2.952
Te^{132}	1.0	.0445	40.406	4.579
I^{132}	1.0	.0445	40.406	4.579
I^{33}	1.0	.0662	60.110	6.811
Xe^{133}	0.994	.0662	59.749	6.770
I^{135}	1.0	.0656	59.565	6.749

(cont'd)

Table G.2. (cont'd)

Isotope	s_i (39 days)	y_i	$\frac{t=0}{\text{curies} \times 10^3/\text{Mw}}$	$\%_{i_{30}}^{39}$
Xe^{135m}	0.3**	0.0656	17.870	2.025
Xe^{135}	1.0	.0656	59.565	6.749
Cs^{137}-Ba^{137m}	0.00224	.0619	0.126	0.014
Ba^{140}-La^{140}	0.8785	.0607	48.419	5.4865
Ce^{141}	0.558	.0590	29.893	3.3873
Ce^{143}	1.0	.0610	55.388	6.276
Pr^{143}	0.861	.0610	47.689	5.404
Ce^{144}-Pr^{144}	0.0915	.0576	4.786	0.5423
Nd^{147}	0.914	.0245	20.333	2.304
Pm^{147}	0.0178	.0245	0.396	0.045
Pm^{149}	1.0	.0125	11.350	1.286
Sm^{151}	0.001014	0.0040	0.004	0
			882.510	99.998

* It is recognized that more figures are given than are significant; this was done for convenience only.

**Branching ratio included in s_i.

TABLE G.3. INITIAL DOSE RATES

Isotope	$\%_{i_{30}}^{39}$	f_{a_i}	$\sum_i E(RBE)N$	D_i^o (39 d)	$7.31\, f_{a_i} \times \sum_i E(RBE)N$
Bone					
Sr^{89}	2.1168	0.22	2.8	0.09532	4.503
Sr^{90}-Y^{90}	0.0233	0.22	5.1	.0019111	8.202
Sr^{91}-Y^{91m}	6.420	0.22	3.309	.3417	5.322
Y^{91}	2.2984	0.14	2.8	.06587	2.866
Zr^{95}	2.368	0.058	0.77	.0077	0.326
Nb^{95m}	0.024	0.12			
Nb^{95}	0.743	0.12	0.33	.0022	0.289
Zr^{97}-Nb^{97m}-Nb^{97}	6.749	0.058	6.10	.1745	2.586
Mo^{99}-Te^{99m}	6.070	2×10^{4}	0.69	.00006	0.001
Sb^{127}	0.206	0.00265	3.20	.00013	0.062
Ba^{140}-La^{140}	5.4865	0.2	4.3	.344936	6.287
Ce^{141}	3.3873	0.1	0.792	.0196	0.579
Ce^{143}	6.276	0.1	1.888	.0866	1.380
Pr^{143}	5.404	0.063	1.6	.0398	0.737
Ce^{144}-Pr^{144}	0.5423	0.1	6.3	.024973	4.605
Nd^{147}	2.304	0.1	1.154	.0194	0.844
Pm^{147}	0.045	0.09	0.34	.0001	0.224
Pm^{149}	1.286	0.09	1.82	.0154	1.197

Table G.3 (cont'd)

Isotope	$\%_{i_{30}}^{39}$	f_{a_i}	$\sum_i E(RBE)N$	D_i^o (39 d)	$170.5\ f_{a_i} \times \sum_i E(RBE)N$
Kidney					
Ru^{103}-Rh^{103m}	1.694	0.01	0.184	0.00533	0.314
Rh^{105}	1.029	0.02	0.33	.01158	1.125
Te^{127m}	0.007	0.02	0.28	.00179	0.955
Te^{127}	0.180	0.02			
Te^{129m}	0.108	0.02	0.89	.03187	3.035
Te^{129}	0.942	0.02			
Te^{131m}-Te^{131}	3.056	0.02	0.755	.0787	2.575
Te^{132}	4.579	0.02	0.112	.0175	0.382
Thyroid					$2557.5\ f_{a_i} \times \sum_i E(RBE)N$
I^{131}	2.952	0.15	0.22	2.4914	84.398
I^{132}	4.579	0.15	0.857	15.0542	328.767
I^{133}	6.811	0.15	0.483	12.6202	185.291
I^{135}	6.749	0.15	0.406	10.5117	155.752
Muscle					$1.71\ f_{a_i} \times \sum_i E(RBE)N$
Cs^{137}-Ba^{139m}	0.014	0.36	0.57	0.0005	0.351
External					
Xe^{135m}	2.025				
Xe^{135}	6.749				
Xe^{133}	6.770				

TABLE G.4. TOTAL DOSES

Isotope	T_i	$T_i/0.693$	D_i^o (39 d)	$(1-e^{-\lambda t})$	$D_i^{70/39}$
Bone					
Sr^{89}	52	75.0	0.09532	1.0	7.149
Sr^{90}-Y^{90}	2700	3895	.0019111	0.9986	7.433
Sr^{91}-Y^{91m}	0.404	0.583	.3417	1.0	0.199
Y^{91}	51	73.6	.06587		4.848
Zr^{95}	48	69.2	.0077		0.533
Nb^{95m}					
Nb^{95}	21	30.3	.0022		0.067
Zr^{97}-Nb^{97m}-Nb^{97}	0.706	1.018	.1745		0.178
Mo^{99}-Te^{99m}	2.8	4.04	.00006		0.0002
Sb^{127}	2.182	3.147	.00013		0.0004
Ba^{140}-La^{140}	12	17.32	.344936		5.974
Ce^{141}	31.04	44.8	.0196		0.878
Ce^{143}	1.372	1.98	.0866		0.171
Pr^{143}	11	15.87	.0398		0.632
Ce^{144}-Pr^{144}	180	259.7	.024973		6.485
Nd^{147}	8.54	12.32	.0194		0.239
Pm^{147}	90.5	130.5	.0001		0.013
Pm^{149}	2.20	3.173	.0154		0.049
					34.849

Table G.4 (cont'd)

Isotope	T_i	$T_i/0.693$	D_i^o (39 d)	$(1-e^{-\lambda t})$	$D_i^{70/39}$
Kidney					
Ru^{103}-Rh^{103m}	13.31	19.2	0.00533		0.102
Rh^{105}	1.5	2.164	.01158		0.025
Te^{127m}	13	18.75	.00179		0.034
Te^{127}					
Te^{129m}	10	14.43	.03187		0.460
Te^{129}					
Te^{131m}	1.154	1.665	.0787		0.131
Te^{132}	2.66	3.84	.0175		0.067
Thyroid					
I^{131}	7.7	11.1	2.4914		27.655
I^{132}	0.1	0.144	15.0542		2.168
I^{133}	0.85	1.226	12.6202		15.472
I^{135}	0.278	0.401	10.5117		4.215
					49.510
Muscle					
Cs^{137}-Ba^{137m}	17	24.52	0.00005		0.0012

$$MPI_{SE_{30}}^{25/70} = \frac{25{,}000}{34.849} = 717\ \mu c; \quad MPTID = \frac{7.17 \times 10^{-4}}{5 \times 10^{-4}} = 1.435\ \text{curies}\cdot\text{sec}/m^3$$*

*Based on 39-day operation and a source strength of 882.5×10^3 curies/Mw.

TABLE G.5. DOSE FROM SIX SELECTED ISOTOPES

Isotope	Curies (39 d) x 10^3/Mw	$\%^{39\ d}_{i_6}$	$D^{o}_{i_6}$	$D^{70/39}_{i_6}$	S_i (39 d)
Sr^{89}	18.68	0.1490	0.6711	50.33*	0.3995
Sr^{90}-Y^{90}	0.2055	0.001639	0.01344	52.36	0.00371
Y^{91}	21.38	0.1705	0.4885	35.95	0.3775
Ba^{140}-La^{140}	48.40	0.3861	2.427	42.04	0.8785
Ce^{141}	31.80	0.2537	0.1469	6.58	0.5935
Ce^{144}-Pr^{144}	4.895	0.0390	0.1796	46.64	0.0936

* In mrem per μc inhaled.

BIBLIOGRAPHY

O. G. Sutton, The Diffusion of Matter from an Explosion, PLP-11 Aug. 14, 1946.

R. L. Myers and J. Z. Holland, A Meteorological Survey of the Oak Ridge Area, ORO-99, Nov. 1953.

W. G. Marley, Health Physics Considerations in a Reactor Accident, R/SAF/UK/3 (no date).

F. T. Binford, ORNL Research Reactor Safeguard Report, ORNL-1794, Vols. I and II, Oct. 7, 1954.

M. M. Mills, A Study of Reactor Hazards, NAA-SR-31 (no date).

Kay Way and E. P. Wigner, Rate of Decay of Fission Products, Phys. Rev. 73, 1318 (1948).

K. Z. Morgan (Chairman), Maximum Permissible Amounts of Radioisotopes in the Human Body and Maximum Permissible Concentration in Air and Water, NBS Handbook No. 52, Mar. 20, 1953.

R. L. Waterfield, Cloud Dosage Calculations, XDC 54-4-12, Apr. 5, 1954.

Wm. B. Cottrell, Reactor Program of the Aircraft Nuclear Propulsion Project, ORNL-1234, June 2, 1954.

A. P. Fraas and A. W. Savolainen, ORNL Aircraft Nuclear Power Plant Designs, ORNL-1721, May 1954.

J. H. Buck and W. B. Cottrell, Aircraft Reactor Experiment Hazards Summary Report, ORNL-1407, Nov. 24, 1952.

J. H. Sterner and Merril Eisenbud, Epidemology of Beryllium Intoxication, AMA Arch. Ind. Hyg. Occup. Med., 4, pp. 123-51, Aug. 1951.

A. D. Ryon and L. P. Twitchell, Vapor Pressure and Related Physical Constants of UF_4, H-5.385.2, July 25, 1947.

A. P. Fraas and M. E. LaVerne, Heat Exchanger Design Charts, ORNL-1330, Dec. 7, 1952.

W. S. Farmer et al., Preliminary Design and Performance of Sodium-to-Air Radiators, ORNL-1509, Dec. 26, 1953.

P. L. Hill, Alkali Metals Area Safety Guide, Y-811, Dec. 13, 1951.

L. Wilson, Building Codes and Other Criteria, GM-127 (AEC), (no date).

A. P. Fraas, Three Reactor-Heat Exchanger-Shield Arrangements for Use with Fused Fluoride Circulating Fuel, ORNL Y-F15-10, June 30, 1952.

O. G. Sutton, Weather, Apr. 1947, p. 108

O. G. Sutton, Proc. Roy. Soc. London 135A, 155 (1932)

W. F. Davidson, The Dispersion and Spreading of Gases and Dusts from Chimneys, Ind. Hyg. Foundation Amer. Trans. Bull. 72013 (1949

F. W. Thomas, TVA, Wilson Dam, Ala. Plume Observation Watts Bar Steam Plant (1952) unpublished manuscript

P. B. Stockdale, Geologic Conditions at the Oak Ridge National (X-10) Area Relevant to the Disposal of Radioactive Waste, ORO-58 (Aug. 1, 1951)

ANP Project Quarterly Progress Reports, ORNL 1816, 1771, 1729, 1692, 1649, 1609, 1556, 1515, 1439

C. E. Winters and A. M. Weinberg, A Report on the Safely Aspects of the Homogeneous Reactor Experiment, ORNL-731 (June 20, 1950)